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CONSIDERATION OF SEVERE ACCIDENTS IN DESIGN OF ADVANCED WWER REACTORS

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Abstract

Severe accident related requirements formulated in General Regulations for Nuclear Power Plant Safety (OPB-88), in Nuclear Safety Regulations for Nuclear Power Stations' Reactor Plants (PBYa RU AS-89) and in other NPP nuclear and radiation guides of the Russian Gosatomnadzor are analyzed. In accordance with these guides analyses of beyond design basis accidents should be performed in the reactor plant design. Categorization of beyond design basis accidents leading to severe accidents should be made on occurrence probability and severity of consequences. Engineered features and measures intended for severe accident management should be provided in reactor plant design. Requirements for severe accident analyses and for development of measures for severe accident management are determined.

Design philosophy and proposed engineered measures for mitigation of severe accidents and decrease of radiation releases are demonstrated using examples of large, VVER-1000 (V-392), and medium size VVER-640 (V-407) reactor plant designs. Mitigation of severe accidents and decrease of radiation releases are supposed to be conducted on basis of consistent realization of the defense in depth concept relating with application of a system of barriers on the path of spreading of ionizing radiation and radoactive materials to the environment and a set of engineered measures protecting these barriers and retaining their effectiveness.

Status of fulfilled by OKB Gidropress and other Russian organizations experimental and analytical investigations of severe accident phenomena supporting design decisions and severe accident management procedures is described. Status of the works on retention of core melt inside the VVER-640 reactor vessel is also characterized.

1. INTRODUCTION

Problems of severe accident consideration in large, VVER-1000 (V-392), and medium size VVER-640 (V-407) reactor plant designs are discussed in the paper. Descriptions of these plants are given in [1] and [2] where main features of design, equipment and main system characteristics of these plants are treated.

2. RUSSIAN NUCLEAR REGULATORY GUIDES REQUIREMENTS RELATING TO SEVERE ACCIDENTS

Requirements relating to beyond design basis accidents leading to severe accidents are treated in the Russian nuclear regulatory guides OPB-88, [3], PBYa RU AS-89, [4] and in Requirements on Contents of Safety Analysis Report for VVER Nuclear Plants, [5].
In accordance with documents, beyond design basis accidents are:

- accidents occurring due to initiator events not considered for design basis accidents,
- accidents accompanied by additional safety system failures as compared to single failure assumed for design basis accidents,
- accidents due to realization of erroneous personnel decisions leading to severe core damage or melting.

These documents prescribe to aim for the evaluated probability value of severe core damage or melting under beyond design basis accidents not to exceed E-5 1/reactor-year.

Moreover, for the sake of exclusion of public evacuation necessity outside the area specified by the regulatory requirements to NPPs siting, it is prescribed to aim for the evaluated probability value of occurrence of the largest specified by these requirements accidental release of radiactive materials not to exceed E-7 1/reactor-year.

If beyond design basis accident consequences analysis reveals that this requirement is not met, additional technical solutions should be provided in the design for management of beyond design basis accidents.

Mitigation of beyond design accident consequences is achieved by accident management and/or by realization of emergency measure programmes to protect population, personnel and environment.

Technical means and measures for beyond design basis accident management should be foreseen by the NPP design.

Any available in good working order means:

- designed for normal operation,
- designed for safety ensurance under design basis accidents

can be utilized to manage beyond desisgn basis accidents.
A special guide should be elaborated to manage beyond design basis accidents in accordance with the design documentation. A sequence to bring into realization of emergency measure programmes of protection of population and personnel in case of occurrence of a beyond design basis accident should be given in this guide.

Development of emergency measure programmes to protect population, personnel and environment and special instructions for personnel to manage these accidents should be realized on base of their analysis.

Analyses of beyond design basis accidents should be done in the NPP design. Moreover, conditions should be given in which fuel melting and/or exceeding of fuel rod destroying specific threshold energy are possible.

In accordance with [5], the list of beyond design basis accidents would be compiled on basis of review of all scenarii leading to exceeding normative personnel and population radiation exposures, radioactivity releases and/or radioactivity contents in the environment established for design basis accidents. Vulnerable features of NPP design, operational procedures and organizational structure of personnel activity that may become as the most probable causes of the conditions above would be determined on base of analysis of event trees.
Possible scenarii of severe accidents are split into groups in which plant process system operation required for accident mitigation is similar. Within each group representative scenarii are found in which four criteria as follows:
- maximum personnel and population radiation exposures
- maximum radioactivity release intensity
- maximum integral radioactivity release
- maximum damage extent of plant system and/or equipment

are fullfilled.

A categorization of beyond design basis accidents according to occurrence probability and to severity of consequences should be given in NPP design.

3. DESIGN DECISIONS FOR SEVERE ACCIDENT PREVENTION AND MANAGEMENT

In accordance with [3], [4], prevention of accidents and large releases outward reactor plant in designs of NPP with VVER-640 (V-407), [1], and VVER-1000 (V-392), [2] is provided due to a systematic realization of a defense in depth concept based on a system of barriers on the way of spread of ionizing radiation and radiactive materials to the environment and a system of technical and organizational measures protecting these barriers.

A set of defense levels are designed which provide protection of the plant and physical barriers from damage and preservation of the population and environment if some barriers would be damaged to some extent.

Passive heat removal systems for core residual heat removal during 24 hour plant blackout are available for provision of core integrity in design and some beyond design basis accidents. In LOCA accidents emergency core cooling system is flooding reactor from high and low pressure hydroaccumulators in succession. In the VVER-640 reactor plant an emergency pool connected by piping with refuelling water storage tank is created during this process.
Opening of the automatic depressurization valves secures core cooling through circulation circuit connecting refuelling water storage tank with reactor. Heat transfer to the atmosphere is realized with the containment passive heat removal system.

Substantiation of fulfillment of regulatory guides requirements in severe accidents can be divided into 4 work fields relating to provision of integrity of the following parts:

- reactor core
- reactor vessel
- reactor dry well and core catcher
- containment structure.

On this basis, development of the designs considered is conducted in direction of elaboration of devices aiming at:

- core damage prevention with aid of active and passive safety systems
- mitigation of corium-reactor vessel and corium-concrete well interaction
- provision of outer reactor vessel cooling (for VVER-640)
- corium catching in concrete well.

Concept of VVER-640 enhanced safety is based on corium retention inside reactor vessel. Possibility of outer reactor vessel cooling is ensured. A deflecting shell with a central hole is installed under the vessel. This shell directs the coolant flow along the bottom generatrices. Riser and lower channels are provided in the concrete well.

Works [6], [7],[8], [9] are devoted to investigation of possibility of core melt retention inside reactor vessel under conditions of postulated core meltdown. These works confirm theoretical possibility of core melt retention inside the VVER-640 reactor vessel.

Beyond design basis accident management is one of the defense levels and comprises a system of actions oriented to prevent accident progression to severe accidents or to mitigate severe accidents if they occur. Questions of VVER severe accident mitigation strategy are considered in [10].

Distinctive degrees of severity of possible accident progression based on condition of the defense barriers:

- reaching the maximum design basis fuel rod damage limit,
- a further core damage and/or melting,
- reactor vessel meltdown and/or destruction,
- containment damage

are considered when developing procedures for beyond design basis accident management.

Particular safety goals are formulated as related to each degree of severity and safety functions are determined of which fulfilment is required to attain these goals.

In case of failure to attain the goals for some degree of severity, the actions are determined which are oriented to delay a further damage progression on base of necessity of attainment of the safety goals for the next degrees of severity.

Measures for severe accident management are oriented to:
- prevention of core damage,
- retainment of the damaged or melted core if any inside the reactor vessel,
- preservation of containment integrity,
- limitation of radioactive releases into the environment.

The following means:
- any available efficient technical means designed for normal operation,
- systems designed for safety ensurance under design basis accidents

can be utilized for accident management to prevent the situation from progressing to a core melt.

In fact, the risk of core melt occurrence may be thought of as being directlly related to violation of two critical safety functions: reactor subcriticality function and reactor core cooling function.

However, fulfillment of primary boundary integrity function, primary coolant inventory function and ultimate heat sink function creates necessary prerequisites for fulfillment of the two functions above.

Reactor subcriticality provision implies:

- a timely reactor trip,
- provision of a sufficient subcriticality margin after reactor trip,
- prevention of inadmissable reactivity variations

A timely reactor trip for the VVER-640 and VVER-1000 is provided due to appropriate trip insrtrumentation and trip set-point adequacy.

A sufficient subcriticality margin after reactor trip is provided due to an adequate worh of shutdown rods and of emergency boron injection systems

Prevention of inadmissible reactivity variations is provided by timely reactor trip and boron injection into the reactor:

- from hydroaccumulators with actuation, if necessary, of the automatic primary depressurization system (VVER-640),
- from boron injection pumps and hydroaccumulators of the emergency core cooling system (VVER-1000),
- from the quick boron supply system in case of a reactor trip failure (VVER-1000),
- from the normal makeup system in conjunction with normal boron control system if they are efficient.

Fulfillment of the primary boundary integrity function or primary boundary damage limitation is provided by prevention of inadmissible thermal and mechanical loads on the primary boundary.

Prevention of inadmissible thermal and mechanical loads on the primary boundary in the course of accident management should be ensured on base of strict realization of operational and emergency procedures.

Primary coolant inventory is provided by the same systems as for reactor subcriticality provision.

Means for provision of the ultimate heat sink depend to a large extent on the concrete accident scenario.

Actual procedures of severe accident management at NPPs with advanced VVER are expected to be based on:

- Probability risk assessments defining scenarii with maximum contribution to core damage risk and revealing necessary mesures for accident management.

- Beyond design basis accident analyses necessary for understanding severe accident consequences and effectiveness of accident management measures.

- Instrumentation providing continuous parameter measurement featuring the unit critical safety functions mentioned above like reactor reactivity, hydrogen concentration in containment, contents of radiactive materials in primary and secondary coolant, outside containment radiation level etc.

- Systems giving to the operational personnel computerized information on condition of all safety and safety-related systems

4. EXPERIMENTAL AND ANALYTICAL INVESTIGATIONS OF SEVERE ACCIDENTS

4.1 Experimental investigations of severe accidents

The research and development works oriented to provision of measures and creature of means for prevention or mitigation of severe accidents leading to core damage beyond the limits prescribed for design basis accidents are being performed under special programmes. A systematic solution of design tasks is realized under these programmes on base of carrying out of computational, theoretical and experimental investigations.
Many scientific and research institutes jointly with OKB Gidropress participate in these programmes.

The approach adopted to design works for fulfillment of regulatory guide requirements in severe accidents determines contents of experimental and analytical investigations necessary for substantiation of design solutions. Experiments are aimed at process investigation of:
- core destruction
- corium transportation inside the core structure
- steam generation and steam explosions as a result of corium-water interaction
- in core melt bath
- heat removal from vessel outer surface
- corium-coolant, corium - metal of rector vessel and corium- concrete interactions.

The following experiments are planned to be performed in the first priority series:
- integral tests of fuel assembly destruction on PARAMETER test facility (Louch Scientific and Industial Association)
- investigation of radiactive fission product release following seal failure of fuel rods (Obninsk Institute of Physics and Power Engineering)
- investigation of radiactive fission product release during corium melting on RASPLAV-2 test facility (Sosnovy Bor Science and Research Institute of Technology).

For investigation of reactor vessel behavior experiments are planned on:
- thermal and mechanical properties of the vessel steel (Obninsk Institute of Physics and Power Engineering)
- boiling crisis on the vessel outer surface on PETLA test facility (Sosnovy Bor Science and Research Institute of Technology) and OKB Gidropress test facility
- water behavior on the surface of corium melt (Sosnovy Bor Science and Research Institute of Technology, Obninsk Institute of Physics and Power Engineering)
- physical and chemical processes in corium and its interaction with the vessel steel on RASPLAV programme (Kurchatov Institute Russian Research Center) and on RASPLAV-2 test facility (Sosnovy Bor Science and Research Institute of Technology, Obninsk Institute of Physics and Power Engineering)
- processes at core melt dropping into water on LAVA test facility (Louch Scientific and Industial Association), on BAK, VULCAN, TVMT test facilities (Obninsk Institute of Physics and Power Engineering).

Tests on RASPLAV-2 test facility are planned for investigation of fission products and gases release in concrete- core melt - water composition.

4.2. Analytical investigations of severe accidents

Analytical investigations of severe accident processes are carried out using integral (considering reactor plant as a whole) and specific (for investigation of separate structures) computer codes. National and foreign computer codes are utilized for this goal.

Processes occurring during core drainage including seal failure of fuel rods, their destruction and melting taking into account of physical and chemical reactions, hydrogen and radionuclides release are analyzed with aid of RAPTA-SFD (Science and Research Institute of Inorganic Materials) and SVECHA (IBRAE) computer codes. In addition, SVECHA allows to evaluate core debris relocation into lower reactor plenum.

Analysis of heat transfer in "corium bath- vessel wall - water system" for substantiation of the conceipt of corium retention inside VVER-640 reactor vessel was performed [6] using MELVES computer code (Sosnovy Bor Science and Research Institute of Technology).

At elaboration stage are NARAL (Elektrogorsk Scientific Research Center) and TF PWR (Obninsk Institute of Physics and Power Engineering) computer codes that will allow to calculate processes of core melt bath formation taking into account convection, stratification and heat transfer .

Also elaborated is HITEF code (OKB Gidropress) that will allow to calculate heat up, deformation, melting and destruction of the bottom and cylindrical part of reactor vessel occurring as a result of its physical and chemical interaction with core melt.

RASPLAV computer code (IBRAE) allows to calculate processe starting from core melt bath formation up to vessel destruction.

DINCOR elaborated computer code will cover all the processes mentionned above taking into account possibility of non-symmetric corium formation and vessel bottom destructure.

Investigation of steam explosions during supplying water into reactor on stage of core degradation will be performed with aid of elaborated VAPEX computer code (Elektrogorsk Scientific Research Center).

Foreign computer codes like MELCOR, RELAP-SCDAP (USA), ATHLET-CD (Germany) and other are utilized for testing of ready-made or elaborated computer codes .
As an example participation of OKB Gidropress specialists from 1993 jointly with specialists of other scientific and research and project organisations of the Russian Federation in fulfillment of the International Standard Problem ISP36 [11].
This task is an investigation conducted with the aim of comparison of experimental and computational results of behavour of a VVER assembly model under conditions of an early core degradation stage.

The results of CORA-W2 experiment conducted in Karlsruhe nuclear research center in Germany were used as experimental results. Computations of the Russian specialists had been performed using RAPTA-SFD, MELCOR 1.8.2, ATHLET-CD MOD 1.1, ICARE2 MOD 1.0, SCDAP/RELAP5/MOD 3.1 computer codes.

It is noted in the report [11] on this work that:

- the utilized computer codes describe correctly the test bundle behaviour as a whole,
- a further improvement of the codes is required from the viewpoint of a more realistic models of fuel clad destruction, material interaction etc.,
- further efforts of the participants of the standard problem should be oriented also to study of a later stage of core destruction.

Works on computational and experimental justification are related with conduction of additional expensive investigations The complexity of investigations and generality of investigation goals for VVERs and PWRs makes international cooperation for fulfillment of these works advisable.

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