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VVER-SPECIFIC FEATURES REGARDING CORE DEGRADATION

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ORGANISATION FOR ECONOMIC CO-OPERATION AND DEVELOPMENT

Pursuant to Article I of the Convention signed in Paris on 14th December 1960, and which came into force on 30th September 1961, the Organisation for Economic Co-operation and Development (OECD) shall promote policies designed:

- to achieve the highest sustainable economic growth and employment and a rising standard of living in Member countries, while maintaining financial stability, and thus to contribute to the development of the world economy;
- to contribute to sound economic expansion in Member as well as non-member countries in the process of economic development; and
- to contribute to the expansion of world trade on a multilateral. non-discriminatory basis in accordance with international obligations.

The original Member countries of the OECD are Austria, Belgium, Canada, Denmark, France, Germany, Greece, Iceland, Ireland, Italy, Luxembourg, the Netherlands, Norway, Portugal, Spain, Sweden, Switzerland, Turkey, the United Kingdom and the United States. The following countries became Members subsequently through accession at the dates indicated hereafter: Japan (28th April 1964), Finland (28th January 1969), Australia (7th June 1971), New Zealand (29th May 1973), Mexico (18th May 1994), the Czech Republic (21st December 1995), Hungary (7th May 1996), Poland (22nd November 1996) and the Republic of Korea (12th December 1996). The Commission of the European Communities takes part in the work of the OECD (Article 13 of the OECD Convention).

NUCLEAR ENERGY AGENCY

The OECD Nuclear Energy Agency (NEA) was established on 1st February 1958 under the name of the OEEC European Nuclear Energy Agency. It received its present designation on 20th April 1972, when Japan became its first non-European full Member. NEA membership today consists of all OECD Member countries except New Zealant and Poland. The Commission of the European Communities takes part in the work of the Agency.

The primary objective of the NEA is to promote co-operation among the governments of its participating countries in furthering the development of nuclear power as a safe, environmentally acceptable and economic energy source.

This is achieved by:

- *encouraging harmonization of national regulatory policies and practices, with particular reference to the safety of nuclear installations, protection of man against ionising radiation and preservation of the environment, radioactive waste management, and nuclear third party liability and insurance;*
- *assessing the contribution of nuclear power to the overall energy supply by keeping under review the technical and economic aspects of nuclear power growth and forecasting demand and supply for the different phases of the nuclear fuel cycle;*
- *developing exchanges of scientific and technical information particularly through participation in common services;*
- *setting up international research and development programmes and joint undertakings.*

In these and related tasks, the NEA works in close collaboration with the International Atomic Energy Agency in Vienna, with which it has concluded a Co-operation Agreement, as well as with other international organisations in the nuclear field.

COMMITTEE ON THE SAFETY OF NUCLEAR INSTALLATIONS

The NEA Committee on the Safety of Nuclear Installations (CSNI) is an international committee made up of scientists and engineers. It was set up in 1973 to develop and co-ordinate the activities of the Nuclear Energy Agency concerning the technical aspects of the design, construction and operation of nuclear installations insofar as they affect the safety of such installations. The Committee's purpose is to foster international co-operation in nuclear safety amongst the OECD Member countries.

CSNI constitutes a forum for the exchange of technical information and for collaboration between organisations which can contribute, from their respective backgrounds in research, development, engineering or regulation, to these activities and to the definition of its programme of work. It also reviews the state of knowledge on selected topics of nuclear safety technology and safety assessment, including operating experience. It initiates and conducts programmes identified by these reviews and assessments in order to overcome discrepancies, develop improvements and reach international consensus in different projects and International Standard Problems, and assists in the feedback of the results to participating organisations. Full use is also made of traditional methods of co-operation, such as information exchanges, establishment of working groups and organisation of conferences and specialist meeting.

The greater part of CSNI's current programme of work is concerned with safety technology of water reactors. The principal areas covered are operating experience and the human factor, reactor coolant system behaviour, various aspects of reactor component integrity, the phenomenology of radioactive releases in reactor accidents and their confinement, containment performance, risk assessment and severe accidents. The Committee also studies the safety of the fuel cycle, conducts periodic surveys of reactor safety research programmes and operates an international mechanism for exchanging reports on nuclear power plant incidents.

In implementing its programme, CSNI establishes co-operative mechanisms with NEA's Committee on Nuclear Regulatory Activities (CNRA), responsible for the activities of the Agency concerning the regulation, licensing and inspection of nuclear installations with regard to safety. It also co-operates with NEA's Committee on Radiation Protection and Public Health and NEA's Radioactive Waste Management Committee on matters of common interest.

Contents

1. Introduction

The objective of this report is to compare VVER reactors to pressurised water reactors (PWRs) of Western design from the point of view of core degradation phenomena using the terminology which was applied to the systematisation of severe accident phenomena in earlier CSNI reports. In the following the acronym “PWR” is used for a PWR of Western design.

The basic design features are described and the most important parameters are summarised in order to identify the differences between the two reactor types. In some specific cases the comparison shows more similarities with boiling water reactors (BWRs) than with PWRs. The known VVER experimental support is also summarised.

RBMKs are not included in this report, as this reactor type is not operated in OECD countries, furthermore its design is completely different from those of VVERs and PWRs.

The scope of this report is limited to in-vessel severe fuel damage phenomena. Neither thermal hydraulic processes involving no core degradation, nor containment phenomena, are discussed in detail.

2. Design

The VVER (water-cooled water-moderated power reactor) is a pressurised light water reactor of Soviet design. The first prototypes were constructed in the sixties. Later the serial VVER-440 and VVER-1000 types were designed and built in the Soviet Union, in several Eastern European countries and in Finland.

The first series VVER-440/230 had only limited emergency injection systems and no containment. In the later VVER-440/213 NPPs the safety systems were improved, passive components were included and the confinement function was covered by special building designs. The most sophisticated design, the VVER-1000, has many common features with Western PWRs, including a full pressure containment and cluster type control rods. The VVER-440 type reactors were not standardised and there are significant differences between the units of same series (e.g. the Loviisa units have Western primary pumps, ice condenser and containment).

More than forty VVER units are in operation today. In OECD countries the following nuclear power plants have VVER units:

Dukovany (Czech Republic)	4 VVER-440/213 units
Temelin (Czech Republic)	2 VVER-1000 units (under construction)
Loviisa (Finland)	2 VVER-440/213 units
Paks (Hungary)	4 VVER-440/213 units

Five VVER-440 units at the Greifswald (Germany) power plant were decommissioned in 1990.

Some of the above VVER units have very good operational experience and are listed among the top ten reactors regarding lifetime load factors. Occupational dose levels are lower than in many Western plants, apparently due to selection of materials, high capacity systems for purifying primary coolant and water chemistry control. The safety analyses carried out in the above countries showed that VVERs generally can meet the safety criteria used for PWRs.

2.1 General

The VVER-440/230 type was developed in the Soviet Union between 1956 and 1970. The heat from the primary circuit is removed in six coolant loops using horizontal steam generators, which is probably the most specific feature of all VVERs. The secondary side of the steam generators contains large water volumes covering the heat transfer tubes, which are horizontally placed between the hot and cold collectors. The steam generators can be isolated by main gate valves (MGV) located in both the cold and hot legs. The MGVs isolate the primary coolant pumps as well, however the connecting tubes of the pressuriser and emergency core cooling (ECC) injection are not isolated. The MGVs allow the operators to take one or more of the six loops out of service in case of emergency (e.g. primary to secondary leaks). The primary coolant pumps of this design have wet motors and little inertia, which results in fast coastdown. Due to the geometrical arrangement of steam generators and cold leg connections to the reactor vessel there are loop seals in both the hot and cold legs. The accident localisation system - which serves as a reactor confinement - was designed to handle only one 100 mm pipe rupture. If a larger LOCA happens, this system vents directly to the environment through safety flaps, which open at 0.2-0.5 bar overpressure. The confinement has small volume, poor leak-tightness and poor hydrogen mitigation. The VVER-440/230 has no emergency core-cooling systems and auxiliary feedwater systems similar to those required in Western plants. The plant instrumentation and control, safety systems, fire-protection systems, quality of materials, construction, operating procedures and personnel training are below Western standards. The VVER-440 core has low power density, which is a positive feature regarding severe accident progression.

The VVER-440/213 was designed between 1970 and 1980. The development of this design coincided with first uniform safety requirements drawn up by the Soviet designers. The primary coolant pumps were improved, and the design of six coolant loops with horizontal steam generators was kept. The emergency core cooling systems were extended and an auxiliary feedwater system was added. Hydroaccumulators were connected to the downcomer and to the upper plenum, which makes possible reflooding of the core from both top and bottom. VVER-440/213 NPPs have three trains of emergency core cooling and two trains of the emergency feedwater system. The hermetic rooms of VVER-440/213 reactors are designed to stand high overpressure (1.5 bar) and are connected to the “bubbler-condenser” tower, which consists of large trays filled with water. In case of accidents the air-steam mixture enters the water volume of trays through some labyrinths and the steam condenses in water. The trays in large break loss-of coolant accidents (LBLOCAs) act as a passive sprinkler system. The hermetic rooms still have poor leak-tightness. In the VVER-440/213 the plant components were standardised making possible incremental improvements and backfits. Concerning plant instrumentation and controls, significant variations exist among countries operating this reactor type. Separation of plant safety systems and fire protection were improved over those of the VVER-440/230, but generally are below Western standards. There are major variations in emergency procedures, operator training and operational safety among the plants operated in different countries. Some countries added safety features to their VVER-440/213 nuclear power plants.

The VVER-1000 was designed between 1975 and 1985. It was based on the requirements of a new Soviet nuclear standard that incorporated international practice in the area of plant safety. The power density of the VVER-1000 is comparable to that of PWRs of similar total power. The VVER-1000 has larger steam generators than those of the VVER-440 (11000 tubes compared with 5000 tubes) and has only four primary loops. Loop seals exist only in the cold legs and there are no MGVs in the loops. The VVER-1000 has a full pressure (4 bar) steel-lined, pre-stressed large-volume containment structure, which is similar in function to those in Western nuclear power plants. The hexagonal structure of fuel assemblies was kept, however the external shroud was removed and the control rod design was changed. Instrumentation and control systems, operating and emergency procedures vary greatly among operators of VVER-1000 plant.

Summarising the comparison of NPP parameters (Table 1 [1],[2]) it can be concluded that the VVER-1000 is a very similar design to Western PWRs. The VVER-440 has some conservative advantages, however the earlier VVER-440/230 design has only limited safety systems.

Table 1: NPP data

	VVER-440	Westinghouse 2 loop	VVER-1000	Westinghouse 4 loop
Thermal power [MW]	1375	1192	3000	3400
Number of loops	6	2	4	4
Steam-generator	horizontal	vertical	horizontal	vertical
Primary pressure [MPa]	12.5	15.5	15.7	15.5
Core inlet temp. [^{o}C]	267	284	288.11	292.3
Core Δt [^{o}C]	30	29.9	29.9	34.8
Core flowrate [kg/s]	8800	6916	16711	17700
Average power rate [kW/l]	83.	102.6	111.	104.5
Primary coolant mass [t]	169	114	240	264

2.2 Core Materials

The VVER reactor cores are constructed of different materials from those of PWRs.

Probably the most significant difference is the cladding material: Zr1%Nb alloy in VVERs and Zircaloy (ZrSn alloy) in PWRs. The Zr1%Nb alloys have good operational experience at low temperature, where they are more resistant to oxidation than Zircaloy. The VVER fuel cladding at the end of cycle has much thinner oxide layer and takes up much less hydrogen than PWR cladding. For this reason the Zr1%Nb cladding appears to have considerable ductility even at high burnups.

The mechanical behaviour of VVER cladding is similar to that of Zircaloy-4. The alpha-to-beta phase transition takes place at ~100K lower temperature in Zr1%Nb than in Zircaloy and this affects the thermo-mechanical phenomena (such as plastic deformation) taking place between 800-1000^{o}C. The VVER cladding is capable of taking up slightly more hydrogen at high temperature, and the material becomes brittle at lower oxidation rates. In the Kurchatov Institute, Russia a new version of the MATPRO library was developed, which included the basic mechanical properties for Zr1%Nb cladding. [3]

The original VVER spacers were made of stainless steel. This material interacts with the cladding at about 1000°C, resulting in eutectic formation. The new Russian fuel design uses Zr1%Nb alloy for spacer grids.

The absorber part of control assemblies in VVER-440 reactors is made of boron steel. In the original VVER-1000 design, B_4C control rods were used, covered by stainless steel cladding. The Temelin units will use Westinghouse VVANTAGE-6 fuel assemblies with one Zircaloy-4 central instrumentation tube. Fuel rod cladding is fabricated of Zircaloy too. All spacer grids are made of Zircaloy excluding the top and bottom ones which are of Inconel. The Temelin units will use Westinghouse-type absorber rods with B_4C in the upper part and Ag-In-Cd in the lower part, both are in stainless steel tubes. The discrete burnable absorbers consist of rodlets with solid Al_2O_3-B_4C pellets as burnable absorber material in Zircaloy cladding. The Integral Fuel Burnable Absorbers (IFBAs) are made by coating the selected fuel pellets with thin layer of zirconium diboride (ZrB_2).

In reference [4] the VVER core material data are discussed in detail. The high temperature interaction of core components, cladding oxidation and hydrogen uptake are reviewed. It was concluded that the main differences between PWR and VVER cladding concern the Zr1%Nb oxide scale, the hydrogen uptake and the alpha-to-beta Zr phase transition.

Table 2: Core data

	VVER-440	Westinghouse 2 loop	VVER-1000	Westinghouse 4 loop
Core diameter [m]	2.88	2.45	3.16	3.38
Core height [m]	2.42	2.44	3.63	3.66
Number of assemblies	349	121	163	193
Number of rods per assembly	126	179	312	264
Control rod type	control assembly	cluster	cluster	cluster
Fuel mass [t]	42.	31.7	91.8	101.
Absorber material	boron steel	AIC	$B_4C^{\#*}$/AIC*	AIC

#Russian design VVER-1000 fuel
*Temelin data with Westinghouse fuel

2.3 Fuel

The core of VVER reactors is characterised by hexagonal geometry: the fuel assemblies have hexagonal form and the control rods/assemblies are arranged on the basis of hexagonal symmetry.

The fuel rods in a VVER assembly are arranged on triangular lattice. The VVER-440 assemblies are covered by Zr2.5%Nb shrouds. There is a by-pass flow between the external surfaces of assemblies. The cross-flow between assemblies is very limited, since the shroud has very little perforation located at the lower and upper parts. There is no shroud in VVER-1000 assemblies, except for Novovoronezh-5, which was the first unit of this design.

The spacers used to fix the rods in the hexagonal assembly are smaller than PWR spacers and their number is greater. There is no PWR-type debris filter, only a lower tie plate, at the bottom of a VVER assembly.

The core of a VVER-440 contains 312 standard fuel assemblies and 37 control assemblies. The control assembly is twice as long as the standard assembly. Its upper part is a hexagonal boron-steel (2% B, 20% Cr, 16% Ni) absorber. The lower part of control assembly is the “follower”, which consists of fuel rods and which is similar to the normal fuel assembly.

In normal operation most of the control rods are in their upper position: the followers are in the core and the absorbers are above. During a reactor scram each control assembly moves down: the follower goes below the core and the absorber part enters the core from the top.

In original VVER-1000 reactor types the control rods are similar to PWR clusters, but the absorber material is different.

The fresh UO_2 pellets in VVER fuel have central holes, which can affect the fission gas release mechanism. The hole disappears at higher burnups.

In VVER-440 the fuel assemblies with 1.4%, 2.5% and 3.6% enrichment are used in a three-year cycle. Original VVER-1000s are operated with 2.3%, 3.3% and 4.4% enrichment fuel in the first core and 4.4% in equilibrium cores. The Temelin VVER-1000 will operate with 1.3%, 1.7%, 2.49% and 3.67% enriched fuel in the first core and 3.95% enriched fuel in the equilibrium ones [5].

No neutron poison rods are used in VVERs.

Table 3: Fuel data

	VVER-440	Westinghouse 2 loop	VVER-1000	Westinghouse 4 loop
Fuel type	hexagonal with shroud	square	hexagonal	square
Rod diameter [mm]	9.1	10.72	9.144	9.14
Pellet diameter [mm]	7.8	9.29	7.844	7.844
Lattice pitch [mm]	12.2	14.12	12.75	12.60
Diameter of inscribed circle [mm]	5.1	9.25	5.57	8.68
Fuel lattice	triangular	square	triangular	square
Number of spacers	11	8	15#/9*	8

#Russian design VVER-1000 fuel

*Temelin data with Westinghouse fuel

2.4 Reactor Pressure Vessel

In VVER reactors the hot and cold legs are connected to the reactor vessel at different elevations. The hot legs do not penetrate the downcomer as in PWRs, but they have connections at higher elevation to the upper plenum. As a result of this arrangement there is more water above the core, which can slightly delay core uncovery.

The first generation reactor vessels (VVER-440/230) were made of steel with a high level of impurities (P and Cu). The vessel have no internal lining and there are welds in the vessel wall at the elevation of core. There is concern about embrittlement of these welds due to the high neutron fluence. The VVER-440/213 and VVER-1000 reactor vessels have an internal lining to provide better resistance to corrosion. These reactors were made of steel with low level of impurities and have no welds in the core region. The reactor vessels are produced not only in Russia, but in the Czech Republic as well, and the vessels fabricated in the two countries do not have the same quality.

In the lower part of VVER-440 reactor vessels a large volume was constructed to include the guide tubes for control assembly followers. The bottom of the guide tubes was fixed to an additional plate. The large amount of water in the lower plenum provides some reserve to remove decay heat.

The reactor vessel wall of a VVER-440 is thinner than those of PWRs, since the primary pressure is lower. On the one hand this can result in slightly faster melt-through, but on the other hand in thinner walls less thermal stress can be developed. Another difference is that the bottom part of the reactor vessel is as thick as the cylindrical part in VVERs, while in PWRs that part has much thinner wall than the vertical cylindrical part.

Both VVER-440s and VVER-1000s have an elliptical (or more precisely torispherical) bottom head, which can play a role concerning molten pool formation phenomena in the lower plenum. It can be supposed that debris accumulated in the lower head will be high enough to reach the cylindrical part of the vessel.

The only operating reactor in the world, which has the in-vessel retention of corium as an approved severe accident management measure is the Loviisa plant (VVER-440/213). The approach selected takes advantage of the unique features of the plant such as low power density, a reactor vessel without penetrations and ice-condenser, which ensures a flooded cavity.

Table 4: Reactor vessel data

	VVER-440	Westinghouse 2 loop	VVER-1000	Westinghouse 4 loop
Internal diameter* [m]	3.84	3.327	4.136	4.19
Height [m]	12.0	10.6	13.531	13.36
Bottom head form	elliptical	hemispherical	elliptical	hemispherical
Wall thickness* [m]	0.15	0.181	0.1925	0.216

* in core region

3. Core degradation in VVERs

In-vessel core degradation for PWRs and BWRs was discussed in detail in reference [6]. The review of different PWRs showed that there are some differences in the reactor coolant system (RCS) design, but the arrangements of fuel rods, spacer grids, control rods and guide tubes are nearly identical. For this reason core degradation processes are similar in all PWRs.

The differences between a VVER and a PWR reactor core design can result in some differences in the core degradation processes. The objective of this chapter is to discuss these differences using the terminology defined in [6].

3.1 Initial events

The dominant accident sequences starting from normal operation and leading to a core melt in VVER-440s were determined in the AGNES project [7],[8] for the Paks NPP:

- Feedwater line and collector rupture;
- Main steam collector and steamline rupture;
- Loss of power;
- LOCA accidents;
- Unplanned reactor trip

A Russian analysis [9] indicated that in VVER-1000s the following initial events can lead to core melt:

- Loss of offsite power;
- Loss of heat removal in the secondary side;
- Feedwater and steamline rupture;
- LOCA accidents

The Temelin VVER-1000 PSA studies [10], [11] showed that the following five most important initial events can lead to core melt:

- SG Header Cover Leakage;
- SG Tube Leakage;
- Large LOCA;
- Small LOCA;
- ATWS

Considering the above lists and lists of sequences defined for PWRs and BWRs it can be concluded that there are no new initial events in VVERs leading to core melt except the leakage or rupture of SG collector, which generally results in larger break sizes than the SG tube rupture. In spite of the differences in the plant designs the dominant sequences tend to be the same in PWRs, BWRs and VVERs. However their frequency and contribution to core melt may be different even for two plants of the same reactor type.

3.2 Accident progression

A severe accident sequence involves a large number of phenomena. The first period is characterised by thermal-hydraulic and neutronic transients leading to core uncovery and core heat-up. The related VVER-specific phenomena are discussed in [12].

The core melt period starts with oxidation, melting, relocation and slumping in the core region. Then the formation of a molten corium pool in the lower plenum and consequent bottom head heat-up may eventually lead to vessel failure.

Accident progression in VVERs in general is similar to that in PWRs. However the differences due to geometry and materials can result in differences in the accident sequence and in the timing of events.

The time from accident initiation up to core uncovery varies from few minutes to several hours. As a result of larger water volumes in the primary and secondary sides this time is longer for similar scenarios in VVERs than in PWRs. Besides initial inventory mass time to core uncovery depends on heat generation, break size, system pressure etc.

After core uncovery the fuel temperature increases due to the low heat transfer to steam. The high temperatures lead in low pressure sequences to the mechanical deformation (ballooning, rupture) of the cladding. The flow blockage development due to ballooning in a VVER fuel assembly may be different from that in a PWR because of the triangular (compared with a square) lattice. The shroud of the VVER-440 assembly prevents cross-flow between assemblies. So the blockage inside an assembly can result not only in the reduction of the flow path, but in the intense heat-up of the upper part of the bundle, which cannot be cooled by cross-flows. The cladding rupture is affected by the mechanical characteristics of the alloys. The alpha-to-beta phase transition in VVER cladding takes place at lower temperature and cladding rupture can happen earlier than in a PWR.

The high temperature interaction of steam and cladding results in oxidation and hydrogen generation. The exothermic chemical reaction produces a thermal power, which may be comparable to decay heat. Some of the hydrogen produced is absorbed in the Zr alloy. The Zr1%Nb cladding used in VVERs can take up slightly more hydrogen than Zircaloy. The presence of a Zr2.5%Nb shroud in the VVER-440 designs creates an additional surface which will participate in the oxidation/hydrogen generation reaction.

The early phase of core degradation is initiated as a result of eutectic reactions of core materials at temperatures well below the melting points of fuel and cladding. These reactions in VVERs involve control rods, cladding and spacers. The low-temperature interactions generate metallic melts, which are able to dissolve chemically other structures in the core.

With increasing temperature the first reaction is the interaction between cladding and stainless steel spacer, taking place at about 1200 ^{o}C. Another low-temperature reaction in a VVER-1000 takes place between the absorber material B_4C and its stainless steel cladding. The same reaction happens in BWRs, which use the same absorber material. The molten materials flow downward in the core until they reach cooler regions, where the melts solidify and form partial blockages between the flow channels of the fuel rods. In a VVER-440 the melting of the boron steel part of the control assemblies may be the next step of core degradation. The molten steel flows down to the follower part, interacts with the Zr1%Nb cladding and the formation of blockages accelerates heat-up. Melting of core components and break-up of fuel rods lead to the loss of rod-like geometry and to subsequent debris formation.

The UO_2 pellets can be liquefied by dissolution by molten Zr1%Nb or by the interaction between UO_2 and ZrO_2 below their melting points. As a result of these processes ceramic melt can be formed with a high freezing temperature (but lower than those of the original constituents). The low thermal conductivity of ceramic material and the slow heat transfer from it leads to the formation of a molten pool with a ceramic crust.

Steam starvation conditions play an important role in the oxidation within the molten pool. The large water volumes in VVER reactors (especially the large lower plenum of a VVER-440) and also the higher H_2 uptake by Zr1%Nb cladding lead to the conclusion that the hydrogen-to-steam partial pressure will be lower in these reactors than in PWRs. This results in gas conditions which are less reducing.

The late phase core damage phenomena are related to crust failure and melt relocation to the lower plenum. The uncertainties for these phenomena are greater than for the early phase even for PWRs.

If the accident is unmitigated, the molten material will relocate into the lower regions of the core, either as a result of crust failure or of pool overflow from over the top of the crust. The lower core support structures intercept and redirect the melt streams.

In a VVER fuel assembly there is no PWR-like debris filter at the bottom of the assembly. The role of such filters is not known very well in core slumping conditions, however it is supposed that the VVER assembly lower tie plate lets the melt pass through more easily than the labyrinth of a debris filter.

The similarity of core support structures in PWR and VVER-1000 allows the assumption that the relocation process will take place in similar manner.

In the case of a VVER-440 two different relocation paths to the lower plenum can be distinguished: one through the fuel assemblies and lower tie plate, another through control assemblies. The control assembly guide tubes and the guide tube plate create an additional barrier and delay the melt relocation to the bottom head. The power generated in the lower part of control assemblies, which are located below the core, can result in some acceleration of the failure of the lower tie plate. The relocation to the lower plenum is limited due to the localised nature of the breach in the ceramic crust. If the frozen material in the lower part of the assemblies creates complete blockage, then the most likely relocation mechanism will be the side penetration through core barrel.

The main processes in the last period of in-vessel core degradation are molten pool formation in the lower plenum and interaction with the vessel wall leading to vessel failure. The pool thermal hydraulics, crust behaviour and gap formation are affected by the geometry of the bottom head. The corium will be located in the hemispherical bottom head in the PWRs. In VVERs the corium can fill up the elliptical bottom and extend to the cylindrical part of reactor vessel. The bottom wall of VVERs is as thick as the vertical part of the vessel. The bottom head penetrations can accelerate local vessel failure, but VVERs have no such penetrations.

3.3 Phenomena

The previous chapter summarised the main events of in-vessel core degradation. Here the related phenomena will be reviewed and the VVER-specific features will be discussed in more detail.

The identification of phenomena is based on the In-Vessel Core Degradation Code Validation Matrix [6], but some new items have been added as well.

I. Fission and decay heat

Fission and decay heat determine the specific heat source in the fuel which is an important parameter for the heat-up rate as long as chemical reactions are not dominant.

I.1 Burnup	Burn-up is proportional to the generated power and the operating time. It is related to the total mass of uranium. The generated power and the operation cycle in VVERs are similar to those in PWRs. The VVER cladding at high burnup is less oxidised than Zircaloy and this fact may result in different failure mechanisms and mechanical behaviour in the two reactor types.
I.2 Decay time	No VVER-specific phenomena. (Decay time is the time after reactor shutdown)
I.3 Recriticality	As long as the whole or a part of the reactor core remains in its original geometry, recriticality is possible due to boron dilution or absorber separation from the fuel.
I.3.1 Boron dilution	In a VVER-440 the main gate valves can isolate the loops and create water plugs, which might induce recriticality.
I.3.2 Absorber-fuel separation	The absorber materials fail at lower temperatures than fuel rods. After their relocation to lower core regions reflooding with unborated water will induce recriticality. In a VVER-440 the control assembly relocation mechanism is different compared to PWRs. The absorber material is not inside the fuel bundle, but between the assemblies. The

molten boron steel flows down through fewer obstacles than the molten PWR absorber, which flows down in the narrow space between fuel rods through spacers. This phenomenon in a VVER-440 in some ways is similar to that in a BWR.
The control rod relocation in a VVER-1000 is affected by the triangular lattice and by the smaller size and greater number of spacers.
The primary water sources in VVER-440s have sufficient boron content, that reduces the possibility of recriticality.

II. Fluid state

The fluid is on one hand the dominant heat sink in water reactors. On the other hand its oxygen potential might lead to exothermic chemical reactions.

II.1 System pressure	No VVER-specific phenomena (The pressure conditions expected during accidents in VVERs are similar to those in PWRs.)
II.2 Core uncovery	Core uncovery depends on the boundary conditions of the cooling system. The connection of hot and cold legs to the reactor vessel at different elevations and the presence of loop seals in VVERs can have an effect on the timing of uncovery. In the VVER-440/230 the lack of hydroaccumulators can lead to earlier core uncovery.
II.3 Core reflood	Reflood in VVERs can be initiated from both top and bottom owing to the connection of hydroaccumulators. The large lower plenum of the VVER-440 results in a longer time to fill up the reactor vessel. Slight differences are expected in the reflood of VVER bundles due to the hexagonal geometry and shrouded assemblies of the VVER-440.
II.4 Gas composition	No VVER-specific phenomena.(The sources of non-condensable gases are the same in VVERs and PWRs: metal oxidation, hydroaccumulator injection, air ingress)
II.5 Geometry effects	The shape of structures determining the coolant flow path might change during accidents.
II.5.1 Blockage	The hexagonal geometry of assemblies, lack of debris filter, presence of lower tie plate and a different spacer design in VVERs might have an effect on the development of partial or total blockages.

II.5.2 Bypass	In normal operation of VVERs-440 there are bypasses between assembly shrouds and the assembly-core barrel. The bypass flowrate (7-8%) is similar to that in BWRs.
II.5.3 Failure of structures	In the VVER-440 during severe accidents the meltdown of control assemblies can create additional flow paths. The failure of guide tube plate structures below the core is also a VVER-440 specific phenomenon, which might change the flow path.

III. Initial core damage

The initial core damage covers the behaviour of fuel rods, absorber and structural components during the early phase of core degradation including heat transfer, mechanical behaviour, melting and relocation.

III.1 In-vessel heat transfer	The in-vessel heat transfer describes the interaction between core and coolant. Three regimes are possible: boiloff, dry core and quenching.
III.1.1 Boiloff	No VVER-specific phenomena (The steady evaporation of coolant with increasing core uncovery takes place in similar way in VVERs and PWRs.)
III.1.2 Dry core	In a VVER-440 the follower assemblies representing 10% of total fuel load move below the core after scram. When the mixture level drops below the lower core support plate the followers might be still covered and enhance steam production. The main part of steam produced in the followers will flow through the control assemblies to above the core.
III.1.3 Quenching	Slight differences are expected in the reflood of a VVER bundle due to hexagonal geometry and shrouded assemblies of the VVER-440.
III.2 Fuel rod mechanical behaviour	While the fuel maintains rod-like geometry the following main phenomena occur: fuel/cladding contact, clad ballooning, flowering, embrittlement, irradiated fuel effects, fuel and non-fuel dissolution, oxide shell failure.
III.2.1 Fuel-cladding contact	Slight differences are expected between VVERs and PWRs due to clad material, pellet grain size and the central hole in the pellets (except the Temelin VVER-1000 with VVANTAGE 6 Westinghouse fuel).

III.2.2 Ballooning	Ballooning becomes significant at lower temperatures in VVERs than in PWRs due to the earlier alpha-to-beta phase transition in Zr1%Nb than in Zircaloy-4. The lower ductility of VVER cladding material results in smaller deformation.
III.2.3 Flowering	No VVER-specific phenomena.
III.2.4 Embrittlement	The VVER cladding becomes brittle at lower oxidation rates than Zircaloy. However the embrittlement of fuel rods due to cladding-fuel chemical interaction is similar to PWRs.
III.2.5 Irradiated fuel effects	No VVER-specific phenomena (When the fuel is dissolved by molten cladding the same fission products are released in VVERs and PWRs.)
III.2.6 Non-fuel dissolution	The early liquefaction of core components take place at temperatures several hundred degrees below their melting points. The interaction of stainless steel with cladding, and boron steel with cladding are VVER-typical phenomena. The shroud in a VVER-440 is an additional surface for component interactions.
III.2.7 Fuel dissolution	No VVER-specific phenomena (The reaction rate between Zr1%Nb and UO_2 is slightly higher than that between Zircaloy and UO_2 in a PWR.)
III.2.8 Oxide shell failure	No VVER-specific phenomena, however the layered structure of Zr1%Nb oxide shells can have an effect on the failure behaviour.
III.2.9 Absorber assembly behaviour	Light water reactors use the following three kinds of absorber material: Ag-In-Cd alloy (AIC), B_4C, boron steel.
III.2.9.1 Silver-indium-cadmium	No VVER-specific phenomena (The new VVER-1000 in the Czech Republic will use AIC in the lower part of the rods in the new fuel design.)
III.2.9.2 Boron carbide	In the VVER-1000, control rod degradation starts with chemical reactions between B_4C and stainless steel cladding. The interaction rate is similar to that in BWRs.

III.2.9.3 Boron steel	In the VVER-440 the molten part of boron steel control assemblies interacts with the Zr2.5%Nb shroud of the fuel assemblies.
III.2.10 Structural materials	In this section the spacer grids are considered, which may survive chemical reactions.
III.2.10.1 Non-fuel dissolution	The interaction between stainless steel spacer and Zr1%Nb cladding is similar to Zircaloy-Inconel interactions. The smaller VVER spacers could cause reduced local damage on the cladding and so less severe attack on UO_2 pellets.

IV. Oxidation and hydrogen generation

Oxidation of metallic components changes the material composition of the core and leads to additional heat generation and hydrogen production.

IV.1 Material composition

IV.1.1 Zircaloy	The high temperature steam oxidation rate of VVER cladding material is about the same as that of Zircaloy-4. The Zr2.5%Nb shroud of VVER-440 assembly is an additional surface to react with steam and to produce hydrogen.
IV.1.2 Stainless steel	No VVER-specific phenomena (The role of stainless steel oxidation is similar in PWRs and VVERs: less significant than cladding oxidation.)
IV.1.3 Fuel	No VVER-specific phenomena. (Fuel oxidation in steam is similar in VVERs and PWRs.)
IV.1.4 Boron carbide	No VVER-specific phenomena. (B_4C oxidation in VVERs-1000 is similar to PWRs and BWRs.)
IV.2 Oxygen potential	Oxygen potential depends on the fluid state.
IV.2.1 Steam	In the VVER-440 the followers even in dry core can produce steam, so the oxidation and heat-up are less limited than in PWRs. The development of steam starvation conditions needs more time in this reactor. A significant amount of steam produced in the followers can bypass the core through the control assemblies.

IV.2.2 Air	No VVER-specific phenomena. (The presence of air plays a similar role in PWRs and VVERs, but the interaction rates might be different.)
IV.3 Preoxidation	No VVER-specific phenomena (The oxide layer on the cladding surface can delay the reaction with spacer grids in both VVERs and PWRs.)
IV.4 Loss of protective shell	No VVER-specific phenomena. (The protective oxide layer can be lost during the transient in both VVERs and PWRs.)
IV.5 Relocated material	Metal-rich relocating melts and refrozen crust may continue to oxidise and contribute to heat-up and hydrogen production.
IV.5.1 Molten film	No VVER-specific phenomena. (The oxidation of metal-rich relocating melts on their surfaces is probably similar in PWRs and VVERs. The relocated material from a molten VVER-440 absorber assembly might have a lower ceramic fraction and a higher oxidation rate.)
IV.5.2 Crust	No VVER-specific phenomena (Frozen melt can protect cladding from further oxidation in both VVERs and PWRs.)

V. Fission product release

The release of fission products and other materials from the core is strongly coupled with the core behaviour and feeds back to the degradation process.

V.1 Fission product inventory	No VVER-specific phenomena. (The actual fission product inventory can be different for each reactor due to different neutron spectrum, burnup level and reactor operation conditions.)
V.2 Fuel state	Fission product release rates depend strongly on the physical and chemical state of the fuel and fission products.
V.2.1 Solid	The fission products are partitioned in the fuel grain matrix, intergranular bubbles, grain boundary and gap. Their release

	can be affected by the larger grain size and the central hole of the VVER pellets (except in the Temelin VVER-1000).
V.2.2 Liquefied	No VVER-specific phenomena. (Liquefaction of UO_2 can enhance fission product release on grain boundaries in both VVER and PWR.)
V.2.3 Molten	No VVER-specific phenomena. (Several mechanisms of melt formation can appear in both VVERs and PWRs and influence fission product release.)
V.3 Gap condition	No VVER-specific phenomena (The gap acts a reservoir for fission products in VVERs and PWRs, fission products interact with the inner clad surface. Clad failure might be affected by VVER cladding behaviour during ballooning.)
V.4 Oxygen potential	No VVER-specific phenomena. (The oxygen content of the fluid determines the clad and fuel oxidation and has a major effect on the chemical state of the fission products in both VVERs and PWRs.)

VI. Core degradation and melt progression

In the late phase of core degradation the heat-up and material interactions lead to relocation of core materials, formation of debris beds and of molten pools.

VI.1 Metallic interactions	Metallic interactions are eutectic interactions between different metallic core components: fuel rods, spacers, control assemblies.
VI.1.1 Inconel/cladding	Inconel is not used in VVERs (except in the Temelin VVER-1000). The stainless steel spacers liquefy the Zr1%Nb cladding, the molten phase seems to have lower viscosity than the molten phase of PWR cladding/Inconel spacers.
VI.1.2 AIC/cladding	No VVER-specific phenomena. (The VVER-1000 in Czech Republic will use AIC in the lower part of the rods in the new fuel design.)

VI.1.3 B_4C-SS-channel box wall/cladding	No VVER-specific phenomena (A VVER-1000 has no structures like a BWR channel box. In a VVER-440 the molten boron steel absorber material attacks the Zr2.5%Nb assembly shroud in a similar way as B_4C-stainless steel melt attacks the Zircaloy channel box in a BWR.)
VI.2 Limited material relocation	In the relocation of non-fuel material relocation the following three phenomena are concerned: spreading, candling and breakaway.
VI.2.1 Spreading	Spreading is the radial movement of melt caused by a driving force. The shroud of the VVER-440 fuel assembly might hamper this movement.
VI.2.2 Candling	Candling is a gravity-driven downward flow of melts. Candling in the control assembly of a VVER-440 and in a normal fuel assembly will differ due to boron steel melt. The shroud of a VVER-440 fuel assembly might cause some differences in comparison with PWR assemblies.
VI.2.3 Breakaway	Breakaway is the mechanical failure of cladding induced by the same phenomena in PWRs and VVERs: thermal stress, thermal shock, flowering. The different embrittlement characteristics of Zircaloy-4 and Zr1%Nb might affect this process.
VI.2.4 Blockage formation	Mechanical obstruction, a thermal gradient and/or a metallic crust can stop material relocations and result in blockage formation.
VI.2.4.1 Mechanical obstruction	Mechanical obstruction in VVERs is influenced by several factors. The lack of debris filters accelerates relocation in comparison with that in PWRs. However the higher number of spacers in the bundle, the reduced space in a triangular lattice compared with that in a square lattice and the VVER-440 follower structures below the core provide additional obstacles to stop material relocations. The VVER-440 absorber assembly allows relocation bypassing the fuel assembly down to the top of the follower after failure of the shrouds and absorber wall.

VI.2.4.2 Thermal gradient	No VVER-specific phenomena (The effect of a thermal gradient on candling is similar in VVERs and PWRs. If the surface temperature is lower than the liquidus temperature of melt, freezing starts.)
VI.2.4.3 Metallic Crust	No VVER-specific phenomena. (The role of metallic crust is the same in VVERs and PWRs: forms a local blockage and acts as a crucible for melt arriving later.)
VI.3 Debris bed formation	The accumulation of solid particles and/or melts is called debris bed formation The formation of a debris bed is a result of the following phenomena: fuel melting, fuel rod collapse, particulate debris formation, melt pool formation.
VI.3.1 Fuel melting	No VVER-specific phenomena. (Fuel melting takes place in VVERs and PWRs under similar conditions.)
VI.3.2 Fuel rod collapse	No VVER-specific phenomena (The fuel rod might break into pieces and form particulate debris in a similar way in VVERs and PWRs.)
VI.3.3 Particulate debris	No VVER-specific phenomena. (The particulate or solid debris consists of similar pieces and fragments of oxidised cladding and fuel in VVERs and PWRs.)
VI.3.4 Melt pool	A ceramic melt pool is formed by the accumulation of fuel melts or/and oxidised cladding from rods or debris. The upper core structures may also contribute to the melt pool. The accumulation of melt depends on blockage formation which might be different in VVERs compared to PWRs.
VI.4 Debris bed	The debris bed can be characterised by external heat transfer, debris bed thermal hydraulics and melt formation.
VI.4.1 External heat transfer	No VVER-specific phenomena. (The debris bed is cooled by the same phenomena in VVERs and PWRs: conduction, convection and radiation. The relative importance of heat transfer modes depends first of all on the fluid state.)

VI.4.2 Debris thermal hydraulics	No VVER-specific phenomena. (Debris thermal hydraulics can be characterised by the same processes in VVERs and PWRs: critical heat flux, quenching, convection, dry-out, melting. The porosity of debris plays and important role in these processes, but it depends more on the accident sequence considered, than on the reactor type.)
VI.4.3 Melt formation	No VVER-specific phenomena. (The debris overheats and forms a melt pool in similar way in VVERs and PWRs.)
VI.5 Molten pool	The molten pool can be characterised by external heat transfer, pool thermal hydraulics, crust behaviour, crust failure and slumping.
VI.5.1 External heat transfer	No VVER-specific phenomena. (The external heat transfer mechanisms are the same in VVERs and PWRs.)
VI.5.2 Pool thermal hydraulics	No VVER-specific phenomena. (Pool thermal hydraulics is a similarly complex phenomenon in both VVERs and PWRs.)
VI.5.3 Crust behaviour	No VVER-specific phenomena. (Heat flux, crust thickness and stresses vary with location along the boundary of molten pool in both VVERs and PWRs.)
VI.5.4 Slumping	Slumping is a massive melt relocation. The flow path of relocation can be affected by follower assemblies and guide tubes in a VVER-440. The interaction of a melt jet with the guide tube plate might delay slumping. Side penetration through core barrel is also a possible mechanism in VVERs.

VII. Lower plenum molten pool

The last phase of in-vessel core degradation before vessel failure is related to the behaviour of molten pool in the lower plenum.

VII.1 Pool thermal hydraulics	The elliptical form of a VVER bottom head creates a slightly different heat transfer profiles than the hemispherical bottom head of a PWR. Its influence on the convection in the molten pool is very little.

VII.2 Crust behaviour	Crust behaviour might be affected by corium composition, which will be probably different in PWRs and VVERs.
VII.3 Ex-vessel heat transfer	For the Loviisa VVER-440 flooding of the reactor pit occurs by melting of the ice-condensers. Analyses indicate, that the core debris can be sufficiently cooled by ex-vessel heat transfer in this reactor type. For the other VVERs ex-vessel cooling is not foreseen as accident management measure.

Summary on phenomena

The following conclusions were made for the main phenomena groups:

- Fission and decay heat:
  The VVER reactors have large water volumes, which might provide long period before core damage. The high burnup fuel in VVERs is less oxidised than in PWRs and so retains more ductility, which affects its mechanical behaviour and failure mechanism.

- Fluid state:
  The fluid flow paths depend on the geometry of reactors. During blockages and structure failures the VVER design (especially VVER-440 with shrouds on assemblies, control assemblies, followers, structures below the core) creates slightly different cooling conditions than in PWRs.

- Initial core damage:
  The first period of core degradation is similar in VVERs and PWRs, the same phenomena with similar sequence of events are anticipated in both cases. The related experiments showed that some differences are expected due to different interaction rates between core materials and due to the cladding phase transition.

- Oxidation and hydrogen generation:
  The shroud of VVER-440 provides and additional Zr surface to react with steam and to produce hydrogen. The follower part of the control assembly can produce steam even in dry core conditions under the core. These features of VVER-440 create different oxidation conditions and higher hydrogen generation than in PWRs.

- Fission product release:
  The fission product release mechanisms are similar in VVERs and PWRs. The core inventory and the composition of released materials might be different even for the same reactor type.

- Core degradation and melt progression:
  The complex and heterogeneous geometry of VVER-440 influences the steps of core degradation and melt progression. The shroud of fuel assembly may hamper the radial

spreading of molten materials. The melting of the absorber part of control assembly can create large channels for downward flow of melts, while the followers and the structures below the core facilitates blockage formation. For VVER-1000 core degradation and melt progression take place in a similar way as for PWRs, the blockage formation can be affected by the triangular arrangement of fuel rods.

- Lower plenum molten pool:
  In the elliptical bottom head of VVERs the molten pool can extend up to the cylindrical part of the vessel and the corium creates heat flux not only on the bottom of reactor vessel, but on the sidewall as well. The structure and the convective behaviour of the molten pool in the VVER bottom head and in the hemispherical PWR bottom head seems to be similar: only slight differences are expected in the heat transfer profile.

4. Experiments

The experimental investigation of core degradation phenomena for VVER reactors started later and was less extensive than for Western LWRs. The VVER experimental results cover many fewer aspects of high temperature interactions of core materials than the Western LWR database. Furthermore, even the existing experimental data have not been collected and analysed as systematically as has been done for PWRs and BWRs.

A detailed summary is given in reference [4] on VVER-specific material data experiments. Here only a short review will be given of the experiments performed, without discussing the results in detail. The list of VVER-related high temperature experiments is surely not complete.

High temperature VVER experiments have been carried out in Hungary, Russia, the Czech Republic and Germany.

4.1 Separate-effect tests

The Hungarian (KFKI Atomic Energy Research Institute) experimental programme [13],[14] covered the following separate-effect test investigations:

<u>UO_2 pellet/Zr1%Nb cladding interactions</u> in argon and steam atmosphere. The total interaction layer thickness was determined in the 1000-1300 °C temperature range.

<u>Stainless steel spacer/Zr1%Nb cladding interactions</u> in vacuum. The tests were performed in the 1000-1250 °C temperature range without oxide layer formation, and the time of melt-through as a result of eutectic interactions was determined.

<u>Boron steel /Zr1%Nb cladding interaction</u>. The behaviour of absorber material and cladding was investigated between 1080-1235 °C. The results of cladding interactions with UO_2 pellets, spacer and boron steel were converted to Arrhenius equation coefficients and compared with PWR material data. The comparison showed that the interaction rate of VVER cladding material with core components was slightly higher than those of PWRs.

<u>H_2 uptake by Zr1%Nb cladding</u> during steam oxidation. The objective of the tests was to determine the amount of absorbed hydrogen in the 900-1200 °C temperature range. The absolute value of hydrogen uptake by Zr1%Nb was found to be several times higher than that by Zircaloy-4. The design criterion of Zircaloy (<17% local oxidation) was also checked for Zr1%Nb cladding. It was found that the total embrittlement of Zr1%Nb cladding takes place already at 8-10% local oxidation.

<u>Ballooning tests</u>. A large number (~100) of ballooning experiments were performed in the 700-1200 °C temperature range under isothermal conditions with linear pressure increase in an argon

atmosphere with and without cladding pre-oxidation. The experiments showed that the behaviour of VVER cladding is similar in the ballooning process to PWR cladding. However in the range of 800-1000 °C the mechanical (creep) strength of Zr1%Nb cladding is lower than that of Zircaloy-4 due to the difference in the alpha-to-beta phase transition temperature.

In Russia (RRC Kurchatov Institute) experiments were carried out on metallic melt formation and the interaction of the melt with vessel steel of VVER reactor type [15] and also on the VVER fuel behaviour under accidental conditions[16]:

Structural steel X18H10T/Zr1%Nb cladding interaction. The range investigated covered 970-1310 °C, the tests were performed in cylindrical and plate geometries. The kinetic correlations of the interactions were determined. During the simultaneous annealing of Zr1%Nb/X18H10T and the PWR type Zircaloy-4/SS316 material probes it was found that their parameters were very close to each other.

Interaction of vessel steel with the melt of Zr1%Nb/X18H10T. The chemical interaction of VVER reactor vessel steel with molten core material (Zr1%Nb/X18H10T with 25, 50 and 75% steel) was studied between 1040 °C and 1400 °C. The vessel steel residual thickness was determined by post-test examination.

VVER fuel under RIA conditions. Fresh and high burnup fuel rod samples were investigated in research reactors under reactivity initiated accident (RIA) conditions. The observed failure mechanisms - such as ballooning, fragmentation, melting and collapse - depended on the pressure and cooling conditions. The related peak fuel enthalpies were determined. The behaviour of high burnup fuel was very similar to that of fresh fuel because Zr1%Nb retains a large amount of ductility at high burnup.

Czech experiments [4] were also carried out on the basic interactions:

UO_2/Zr1%Nb interaction. Fuel pellets and cladding material disks were pressed to each other, and their interaction was investigated between 1000-1400 °C. The results indicated, that the sequence of reaction layers as well as the total thickness of reaction zone was nearly the same for Zr1%Nb and Zircaloy-4.

Zr1%Nb oxidation by steam. The investigations were carried out in the 600-1200 °C temperature range. They showed that the high temperature oxidation kinetics were comparable with those of Zircaloy-4.

In Finland experiments were carried out to investigate molten pool behaviour in the lower plenum [17].

COPO Experiments. The large scale two-dimensional COPO facility provided experimental data for PWR and VVER geometry under similar conditions. It was found, that the convective heat transfer worked equally well in both geometries.

Comparison of VVER cladding with Zircaloy-4 was investigated in FZ Karlsruhe [18].

Ballooning experiments. Large number of ballooning test were performed with Zircaloy-4 and Zr1%Nb claddings in isobar conditions with increasing temperature.. The results indicated the lower strength of VVER cladding in the temperature range of the phase transition.

The A.A. Bochvar Scientific Research Institute of Inorganic Materials (VNIINM) carries out the basic experiments on core materials for the Russian nuclear industry [19],[20]. Their work includes investigation of material properties and high temperature behaviour of VVER fuel as well.

Acceptance criteria Large number of thermal shock tests were performed in order to check the peak cladding temperature (1200 ^{o}C) and the equivalent cladding reacted criterion (18%) for Zr1%Nb cladding. The experiments with fresh and with high burnup fuels (49.5 MWd/kg) indicated that the VVER fuel does not fail below the above values.

High temperature oxidation of Zr1%Nb cladding was investigated in the temperature range of 550-1300 ^{o}C in steam and hydrogen, steam and air and steam and nitrogen mixtures. The experimental results showed that the addition of air or nitrogen essentially speeded up the interaction process, and the hydrogen content up to 90% caused no change in oxidation kinetics.

The influence of iodine on the mechanical strength of Zr1%Nb cladding was studied in burst tests. 125 mm long specimens were used with constant internal pressure in the range 500-750 ^{o}C. The results indicated that the initial iodine concentration on the internal surface of the cladding had a considerable influence on the rupture deformation of cladding and its effect was decreasing with the increase of temperature.

4.2 Integral tests

Four integral experiments have been devoted up to now to the investigation of real VVER fuel bundles under high temperature conditions: two in the CORA Facility (FZ Karlsruhe) and two in the CODEX Facility (AEKI Budapest). The CORA tests formed the basis for ISP-36.

CORA-W1. Damage initiation and progression of VVER-1000 fuel elements were studied under early phase severe accident conditions. The 19-rod hexagonal bundle included 13 electrically heated rods. The power was linearly increased and steam was injected. Temperature escalation was observed. The upper part of the bundle was strongly damaged and the stainless steel spacer grid melted. The molten material relocated, solidified and formed channel blockages.

CORA-W2. In the VVER-1000 fuel rod bundle one unheated rod was replaced by a B_4C absorber rod. The test procedure was similar to CORA-W1. The big difference between the two tests was the faster movement of the temperature escalation front from the top of the bundle to the bottom caused by the melting and relocation of the B_4C absorber rod. In test CORA-W2 stronger melt relocation and blockage formation were observed.

Both VVER-1000 CORA tests were performed with original Russian fuel rods (Zr1%Nb cladding, stainless steel spacer grids and UO_2 pellets)[21].

CODEX-2. The experiment was performed with a 7-rod VVER-440 fuel bundle. Six rods were electrically heated. The rod bundle was covered by a hexagonal shroud and fixed by three stainless steel spacers. The facility was heated up in an argon atmosphere, then steam was added to the coolant and the electric power was increased. Temperature escalation occurred in the upper part of the bundle which was destroyed during the temperature transient.[22]

CODEX-3. In this experiment a 7-rod VVER bundle was used as well. The test was carried out in two steps. First the facility was heated up, power was increased and at 1200 °C the bundle was quenched by water. This step created an external oxide layer on the cladding surface. In the second step the cooling down by water was initiated at 1500 °C. The upper part of the bundle was damaged, but only a slight temperature escalation was observed.

The VVER integral tests were compared to the other CORA tests, and it was concluded that the high temperature material behaviour of VVER fuel rods is comparable to that of PWR and BWR fuel rods, thus being consistent with the conclusions drawn from the separate-effects material interactions experiments.

5. Conclusions

In this report the design characteristics of Western PWR and VVER reactors have been compared from the point of view of core degradation.

The VVER (water-cooled water-moderated power reactor) is a pressurised light water reactor of Soviet design. It operates on the same principles as a Western PWR reactor and uses similar technological systems. The primary coolant is pressurised water, which heats up in the reactor core and steam is produced on the secondary side of steam generators. The comparison of basic geometrical and technological parameters pointed out some differences between a PWR and a VVER, but it should be noted that differences exist even between two Western PWRs of different design. The VVER reactors are special types of PWRs, the most important design features of which are the horizontal steam generators and the hexagonal core structure.

Similarity between PWR and VVER reactors was found in the comparison of dominant accidents sequences leading to core melt. The accident progression sequence consists of the same steps for VVERs and PWRs. The larger water and metal masses of VVER reactors results for some accident sequences in later core degradation. The unique construction of VVER-440 control assemblies plays a special role during accident progression, having several fuel assemblies below the core and creating a heterogeneous core structure with absorber assemblies. Some events (e.g. B_4C melting in the VVER-1000) are more similar to processes in BWRs. The early phase of core degradation seems to be similar in VVERs and PWRs, however the role of boron steel absorber assembly melting can change the sequence for the VVER-440. Accident progression during the late phase of core degradation can be influenced by the interaction of molten material with lower plenum structures in VVER-440. The mechanism of vessel failure can show some differences due to the VVER bottom heads being elliptical and without penetrations, compared with those of PWRs which are hemispherical and where penetrations are present.

The severe accident phenomena were compared with the help of categories used for PWRs and BWRs. Most of the phenomena were found not to be VVER-specific, which means that the phenomena takes place in a similar way in the reactor types compared. Very few phenomena were found not relevant to VVER (e.g. AIC control rod related ones). When the phenomena was given a VVER-specific character the role of design features was discussed. In some cases there was experimental evidence, however in most of the cases the effect of VVER design could only be estimated. Some phenomena are not known in detail even for Western LWRs, so the comparison can show only some likelihood of differences.

The review of related experiments showed, that the VVER experimental database is not as extensive as that for PWRs and BWRs. A number of separate-effect and integral tests indicated that the behaviour of materials used in VVERs is generally similar to that of PWRs. Some specific areas are not covered by experiments at all and their investigation should be given more attention (e.g. slumping in the guide tubes, relocation behaviour of VVER-440 absorber). There is a need for a systematic evaluation of existing VVER experimental data and for setting up the necessary correlations for numerical models.

The VVER severe accident transients can be simulated with the help of Western SA codes. The material properties embedded in these codes are related to Western materials and they should be substituted by the appropriate VVER ones (e.g. creep rates, oxidation rates, etc.). Without discussing the codes in detail it seems that VVER-1000 can be calculated with these codes without major problems, however in case of VVER-440 modelling problems should be solved first. In some cases BWR code options are more relevant to VVERs than those for PWRs.

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