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TOPICAL REPORT

WORK PACKAGE 5:
Enhanced Safety by Improved Techologies and Materials

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Content

WP 5. Enhanced Safety by Improved Techologies and Materials

1. Introduction

The current reactor designs have been developed in 60's and their operation and safety is in many cases limited by technology development at that time. New materials and new technologies provide a large potential to increase the plant safety, to simplify the plant operation, inspection and repair, and to improve the economics. Development in technology areas outside the nuclear field has in many cases surpassed the practises utilised in nuclear power plants. A benefit should be taken from this when considering new plant designs. The development has been particularly rapid in case of digital systems. The systems have, however, specific research needs when considering nuclear safety applications.

2. Drivers for Innovations

2.1 Utility requirements

The trend in new plant designs is simplification, which is believed to ensure the plant safety, and simultaneously enhance the plant economy. The two components making up the cost of generating power are the capital cost of the plant and the production cost, which is usually broken into fuel cost, and operating and maintenance costs. A key for low capital costs in a nuclear power plant is reduction of the construction time. This can be achieved by effective design methods, new technologies and by system simplification. The design also affects the operation costs. The most important operating parameter to achieve low generation costs is a short outage time, which is essential for a high capacity factor. Designing the plant for easy access and maintenance and for possibility of robotic tooling reduces the outage time, the time needed for repeated in-service inspection, and the exposures of personnel.

The tendency in plant maintenance is to use on a minimum number of active components. This simplifies operation and maintenance but it also places high requirements on the reliability of those components.

2.2 Competitiveness

To be viable in a large scale, any electricity generating system must be safe, reliable and economical. A nuclear power plant will not be built if it is not economical, regardless of the plant safety. Unique to the nuclear generation, a future plant must provide very high protection of investment. This means, that a plant designed and constructed to regulatory requirements can be licensed, that it will not be vulnerable to serious accidents, and that it can be maintained and operated for a manageable and predictable cost. This affects the economic acceptability of innovative technologies, because the investors may not be willing to rely on unproved technology. The designs that have worked well in the past and have been refined and evolved gradually, are usually preferred over novel ideas.

2.3 Licensing requirements

In a nuclear power plant design, safety can not be omitted. The tendency in regulatory requirements for new plants is very low core melt probability and source terms. Some new solutions, digital instrumentation and control for example, have been difficult to license. The problem is to assess quantitatively the high level of reliability required for the safety related software systems.

2.4 Socio-political drivers

The public climate will not tolerate severe accidents, even if they would not lead to environmental releases. Therefore, a large effort must be put on accident prevention.

3. International Trends

3.1 Improved technologies

3.1.1 Improved Instrumentation and Control

Many NPPs in the world have entered a process of modernising their I&C systems. This process has partly been motivated by the difficulty to get spare parts for the old systems and partly by a need for improved functions. In the process one can see that the traditional differentiation between plant control and plant information systems is disappearing. In implementing the new systems at the plants there are basically two strategies which can be followed. Either a complete change of the whole I&C is carried out during an extended outage or the change is implemented in steps over several refuelling outages.

In the exchange of the old I&C systems new technology is brought in. Analogue equipment is exchanged to digital programmable devices. In the modernisation process both a distribution and a centralisation of functions can be seen. Signals are collected through smart transmitter and are routed through computing nodes. In the modern systems the units are interconnected through data highways over which signals are carried in an multiplexed fashion.

Industrial I&C systems have been characterized by rapid changes of technology. Typically a new generation of hardware and software has become available every two years. The second characteristic is the increasing use of computer based systems. For general I&C technology, the nuclear industry is very small, but yet demanding component of the world market. The main issue is how to design, validate and maintain I&C systems to the high levels of safety, reliability and performance required by the nuclear applications. The possibility for software errors is for example bringing in a new kind of common cause error which is very difficult to handle.

In spite of a basic conservativity in applying the new technology there is still much to be gained with the new systems. Signals can easily be duplicated anywhere, there are far larger possibilities to design advanced control algorithms and the man machine interfaces can be made more efficient. The main difficulty with the new systems is to be able to fulfil the licensing requirements of a verifiable reliability for the safety systems implemented with the new technology.

3.1.2 Plant automation

The development towards distributed digital automation systems has been very rapid. There are several trends of development which can be seen today. One is the integration of intelligent functions and stand alone systems into a larger context. The other is the emergence of common platforms for various systems which facilitate their integration. With the new systems plant operation can be simplified to some extent using a higher degree of automation and better operator support systems. Similarly the new automation systems also support maintenance through the use of advanced diagnostic functions. The scheduling and support of maintenance activities have a large potential of being simplified through new automated functions. Inspection activities can be improved through better sensors, robotics and advanced data processing.

Field equipment is also expected to contain embedded computers making them easier to connect to a plant wide information system. Smart components contain their own diagnostic and reporting functions to generate messages which are routed to appropriate receivers. Performance monitoring becomes also easier were components can generate automatic messages telling their states to a function co-ordinating maintenance functions.

3.1.3 Improved man-machine interfaces

The new I&C systems provide the benefit of improved man machine interfaces. Already the Safety Parameter Display Systems (SPDS) are far easier to realise using a computerised platform than using the old analogue technique. Using Visual Display Units (VDU) a far greater flexibility in adapting displays to specific needs connected to operational situations can be achieved than using the old fixed instrument control panels. For the alarm handling advanced filtering schemes can be employed to support the diagnosing function of the operators. Computer supported procedures provide several advantages as compared with paper based procedures. Operator support systems can easily be built using computers for a more efficient and safe operation in various situations.

Additional benefits can be obtained by the computerised man-machine interfaces to implement information support for a large variety of situations. One specific need is connected to low power and shut down operation for which conventional control rooms are poorly adapted. The provision of giving control room operators access to a computerised plan documentation system also opens up interesting possibilities.

The use of a new technology also imposed new needs on the validation process for the control room solutions. It should for instance be ensured that the operator has a good understanding of the new systems and how they should be operated. The restricted window of a VDU also has its inherent limitations which makes it increasingly important to ensure that the information is structured in an optimal way. Various methods for carrying out a control room assessment for the new systems have been proposed.

3.1.4 Expert systems

Expert systems and artificial intelligence (AI) provides interesting opportunities for building new operator support systems. The first generation systems were built simply like a large number of if–then rules which required special computers for their implementation. The systems of today can incorporate a large number of advanced reasoning which gives another paradigm for the construction of the systems. This new way of building AI systems has the benefit of being easier to validate, because they are based on a more restricted set of assumptions.

Expert and AI systems are also closely related to fuzzy reasoning and neural networks. Other interesting applications are intelligent software agents and learning systems which on a time scale of some ten years may develop to a stage where their application in commercial nuclear power plants becomes feasible.

3.1.5 Simulators

Training simulators have already been utilised for a considerable time in the nuclear community. The efficient hardware and software which already today is available at the market brings the possibility of having a full scope training simulator on a desk-top. Simulators can be built by software generators almost directly from the plant design data base.

These developments are due to revolutionise the design and analysis. When a plant design and construction project is started it is likely that the first step will be to build a simulator which is replicated to all designer. The visual display software available also has a provision of making very difficult physical transients understandable for the designer.

3.2 Improved Materials

To realise power plants operating at high temperatures materials with sufficient strength in that temperature regime are needed. Available high temperature heat resistant materials have to be screened with regard to the applicability at the specific temperature, loading and environmental conditions. For new materials presently under development it has to be evaluated whether an application in high temperature reactors is possible and economical. The development has then to be focused on the specific needs for the anticipated application.

Low activation steels are also under further improvement, with the aim of reducing the dose to the personnel.. Several studies are being carried out on the effect of irradiation embrittlement of reactor pressure vessel and core internals materials. These studies will allow to understand better the phenomenon and to take adequate mitigation measures in order to assure the safety of the installation and possibly the extension of their operational life.

The new developments in nuclear technology foresee a more and more extensive use of ceramic materials for components such as the core catcher, the guide tubes and the cladding.

3.2.1 Steels

Reduced activation steels

New steels are currently under development with the aim of improving their mechanical properties and reduce their activation as a consequence of neutron irradiation. Developed in the frame of the researches on thermo-nuclear fusion technology, reduced activation steels could be considered also as structural materials for fission power plants. Advantages can be seen in the reduced dose to the personnel and in easier waste transport and disposal. Preliminary results of tensile tests after irradiation in PWR conditions of several steels of new development were recently obtained at CEA [Averty et al. 1998]. Four classes of steels were tested; the conventional ones will be referred only for comparison.

The development of reduce activation steels foresees the substitution of high long-term activity elements with low long-term activity ones. The former group includes Nickel, molybdenum, niobium, cobalt and aluminum; the latter manganese, chromium, titanium, tungsten and vanadium. In addition very strong concentration limits have to be imposed on many impurity elements with very high long-term activation such as silver, terbium, iridium, samarium europium, bismuth and others. Vanadium alloys obtained with this technique, like for example VA64, exhibit an activation reduced by a factor 2 up to one order of magnitude compared to 316L steel.

Ferritic and martensitic steels have much better thermo-mechanical properties than austenitic ones and moreover a greater radiation damage resistance, therefore there is a tendency to develop new martensitic steels at reduced activation which are now available. CEA has recently presented results on steels of this type characterised by a chromium content in the range 7.5-11%, a tungsten content in the range 0.7-3% and are stabilised with vanadium and tantalum.

Tensile tests have been carried out on LA12LC-(CW) (cold worked) and F82H (tempered) reduced activation steels after irradiation at 325°C up to 3.4 dpa in the OSIRIS reactor. The results obtained show that the steels considered present the same type of behaviour, with the maximum load reached directly after the elastic part and not followed by work hardening. Quite good residual ductility is still present as shown by RA values. The results are collected in the following table.

Table 1. Tensile tests of reduced activation steels after irradiation. T_{irrad} = 325°C

Material	Dose (dpa)	0.2% YS [MPa]	Reduction of Area [%]
LA12LC-(CW)	0	570	71
LA12LC-(CW)	3.4	700	62
F82H	0	460	72
F82H	3.4	720	58

Oxide Dispersion Strenghtened steels

Thermo-mechanical processed super-alloys (e. g. oxide dispersion strengthened ODS) are considered as structural materials in the range of elevated temperatures up to 1250 °C. Possible applications are gas turbine blades and heat exchanger pipes. Iron based ODS materials show good corrosion behaviour also in oxidising atmosphere. Disadvantages are low ductility especially in the high temperature range and the anisotropy due to thermo-mechanical processing.

ODS steels are currently under advanced development. Among them, CEA has recently tested in irradiation conditions a commercial ferritic steel MA 957 in as-extruded conditions and a 9 Cr-1 Mo (EM10) tempered steel reinforced by Y2O3 oxides for application at high temperature and high radiation damage.

Tensile tests results of reduced activation steels after irradiation at 325°C up to 2 dpa recently obtained at CEA are summarized in the following table:

Table 2. Tensile tests of Oxide Dispersion Stengthened steels after irradiation. T_{irrad} = 325°C

Material	Dose (dpa)	0.2% YS [MPa]	UTS [MPa]	Total Elongation [%]	Reduct. Of Area [%]
MA957	0	818	922	17	63
MA957	2	928	973	15.6	57
EM10+Y_2O_3	0	748	851	12	56
EM10+Y_2O_3	2	1076	1103	7.7	41

MA957 ODS grade shows a very good strength/ductility ratio after irradiation at 2 dpa.

Comparison with other steels

The following table reports for comparison the results obtained at CEA concerning other conventional steels in the same irradiation conditions. The two steels considered are respectively:

1) martensitic EM10 type, which at 3.4 dpa still shows capacity of plastic deformation after reaching the maximum load and exhibits a nearly constant RA and TE dependence from the dose level, hence the higher tensile properties stability to radiation respect to the other martensitic steel tested (T91);

2) austenitic 316 LN-(SA), which shows less irradiation hardening than the other austenitic stainless steels tested, i.e. 316-(CW) and especially 304 (SA) for which UTS and YS nearly coincide at 3.4 dpa.

Table 3. Tensile tests of some other steels under same irradiation conditions. T_{irrad} = 325°C

Material	Dose (dpa)	0.2% YS [MPa]	UTS [MPa]	TE [%]	RA [%]
EM10	0	470	560	17	66
EM10	3.4	750	770	12	57
316 LN-(SA)	0	210	475	43	65
316 LN-(SA)	3.4	510	615	25	52

Among the new conception steels considered worthwhile noticing is the very good behaviour of MA957 ODS grade, for which anyway results are available only at 2 dpa, and the interesting post-exposure tensile properties exhibited by the martensitic EM10 at 3.4 dpa.

Further investigation is needed before drawing final conclusions on the radiation stability of the steels under discussion. CEA has planned to extend the irradiation in OSIRIS up to 8 dpa for all the steels considered and therefore new important results will be available in the next two years.

Nitrogen-alloyed austenitic stainless steels

The austenitic stainless steels presently used in nuclear power plants (NPP) for piping and reactor internals, are mainly the types AISI 304 and 316, and in German power plants the stabilized types AISI 321 and 347. In the past, all these steels have shown a certain susceptibility to intergranular stress corrosion cracking (IGSCC). This problem can be overcome by reducing the carbon content to values less than 0.03%. This reduction, however, results in a decrease of the strength values which, in the case of the steels AISI 304 LN and 316 LN, has been (over-) compensated by adding 0.10 to 0.16% of nitrogen: The specified yield strength is increased in both cases from about 180 MPa for the L-variants of these steels to about 280 MPa for the LN-variants.

With respect to austenitic stainless steels, the following points are of interest for the future development of NPP:

- Test the conditions under which the nitrogen-alloyed austenitic stainless steels which are used now, might be susceptible to welding-induced cracking, e.g. hot cracking or ductility dip cracking. This can be done by welding-simulation techniques and should give a good basis to avoid problems during the welding of these steels.
- Examine the question what influence the nitrogen has on the susceptibility of these steels to irradiation assisted stress corrosion cracking (IASCC) which may occur in reactor internals.

Check the applicability and especially the weldability of austenitic stainless steels with nitrogen contents above 0.16%. The advantage of these materials would be
a further increase of yield strength (virtually without a loss in toughness) and an additionally improved corrosion resistance, especially after cold-working.

Core internals

The role of internal structures in a light water reactor is crucial, since they have to ensure the support of the reactor core, maintain the alignment of the fuel assembly and restrain their movements, maintain the separation between the fuel elements and the control rod command mechanism, ensure the flow of coolant under the vessel cover dome, ensure the protection of the reactor against gamma and neutron irradiation, and finally ensure guidance and support of the core instrumentation.

Internals are not considered part of the primary cooling circuit, which constitute the barrier between the reactor medium and the outside, but nevertheless their performance could have an influence on the safety of the primary circuit. Moreover the recently planned reactor life extension would cause an increase in the irradiation fluence which could raise concern about the safety of internals and hence of the primary circuit.

Typical materials constituting internals are 304, 308 and 316 grade steels or welds which deteriorate considerably due to exposure to neutron radiation between 200 and 400°C. Dose levels can reach up to 50 dpa, leading to very low residual ductility and fracture toughness, accompanied by swelling and helium production. The high porosity caused by the dose can moreover lead to a brittle fracture mode called “channel fracture”.

Stabilised steels like 316Ti are also currently used, seeming to have better mechanical strength, less swelling and better resistance to corrosion.

Studies are ongoing about the possibility of designing materials where the formation of dislocation loops is prevented. This can be realised by eliminating point defect and enhancing their probability of recombination by adding titanium in solid solution in presence of carbon.

The segregation of these two elements and possibly the precipitation of TiC should reduce both formation and movements of the dislocation.

Other solutions could foresee the reinforcing of the grain boundaries in order to reduce the irradiation-induced difference between the mechanical strength of the irradiation-hardened grain and that of the grain joining embrittled by helium production.

3.2.2 Ceramic materials

Core Catcher

Application of ceramic materials to in-vessel core retention is under investigation. The so-called "dual strategy" aiming at retaining the corium inside the reactor pressure vessel was designed by taking advantage the study of TMI-2 accident and the results of intensive R&D carried on world-wide on in-vessel corium retention by reactor pressure vessel external cooling.

The material constituting the core catcher must present:

1) a good thermal stability up to very high temperatures (=2500 K);

2) a good stability under irradiation fluence up to 10^{18} n/cm^2;

3) corrosion resistance in the primary fluid containing boric acid and lithia at 300°C

No current single material would be able to fulfil all the severe conditions posed by corium attack, therefore a multi-layer constitution of the crucible wall was recently proposed [Richard et al. 1999]. The innermost layer must be a refractory material able to retain the corium up to very high temperatures and chemically compatible with its constituents. The experience gained with ex-vessel core-catchers led to select ceramic oxides as the most suitable candidate materials.

Among these oxides, materials like ZrO_2 et MgO are the most interesting candidate materials for future core catchers, replacing stainless steel. They proved to be quite satisfactorily resistant to neutron irradiation, with a significant decrease of the mechanical characteristics only for fluences higher than 10^{19} n/cm^2.

The ceramic oxide layer must be protected against the corrosion environment by a stainless steel layer designed also to provide the good mechanical behaviour of the whole wall.

Different scenarios with related assumptions regarding the corium configuration in molten conditions were considered, leading to a multi-layered design foreseeing a layer of magnesium oxide between zirconia and stainless steel. This design should safeguard the oxide to dissolve excessively before reaching the liquidus equilibrium composition. A typical diameter reduction of 5-10% is expected for a 2 m ϕ crucible.

UO_2/ZrO_2 corium meltdown and results of spreading on a SS core catcher are available from FARO experiments [Silverii & Magallon 1998].

Ceramic cladding

The fuel cladding must possess the following properties:

- good thermal conductivity;
- low neutron absorption cross section;

- good mechanical properties up to 300°C and 155 bar which could give rise to creep, including at least 40 thermal cycles, and a dose up to 10^{26} n/m^2;
- good resistance to corrosion;
- thermal expansion coefficient matching the fuel's one

Moreover in case of accident, the cladding should exhibit:

- sufficient structural stability in order to keep the core geometry up to 1200°C,
- good mechanical resistance at high temperature to prevent explosion
- good resistance to thermal shocks during fast temperature rise
- no exothermic reaction with water up to 1370°C.

Zircaloy and Incaloy are the materials currently used as fuel cladding, but studies are carried out on the development of purely ceramic ones as potential alternatives.

Among them Al_2O_3, SiC, carbides, nitrides, ZrO_2 and MgO. Studies on erosion resistance at high temperature have shown that magnesium and zirconium oxides together with alumina have a higher threshold temperature than SiC in presence of water, i.e. up to 1600 against 1400 K.

As for the resistance to corrosion, tests are currently being carried out at CEA. Preliminary results show that pure alumina exhibits a weight loss of only 0.9 mg/dm^2 against the 2 mg/dm^2 of zyrcaloy in the same conditions.

Studies on the behaviour under irradiation of ceramics like MgO, Al_2O_3, $MgAl_2O_4$, $Y_2Al_5O_{12}$, Y_2O_3, Si_3N_4 and SiC have also been carried out at CEA. Fluences considered were up to 2 10^{26} n/m^2 , E>1 MeV, at temperatures up to 1100 K.

Results of swelling measurements ordered by increase of volume show that:

- $MgAl_2O_4$ exhibits less than 0.5%;
- Y_2O_3 and Si_3N_4 between 0.5 and 1.5 %;
- SiC 3%;
- Al_2O_3 in the range 3-6%.

As for mechanical properties, while the flexural strength of Al_2O_3 and Si_3N_4 is practically not affected by irradiation, that of $MgAl_2O_4$ increases up to +30%. To a lesser extent, an increase post-irradiation was found also for the resistance to crack propagation of this material.

Finally it was observed that in non irradiated conditions the thermal diffusivity of alumina is twice that of $MgAl_2O_4$ but the relative magnitude is inverted at high fluences.

For all the above reasons, ceramic oxides and particularly $MgAl_2O_4$ are seriously taken into account as new candidate materials for cladding, but further testing is necessary to fully confirm the results available in literature.

Reinforcement of ZIRCALOY and INCALOY claddings is also studied, consisting in the introduction of a pure zyrconium thin layer on the fuel side. The presence of this layer would improve the resistance to the stress caused by the pellet-cladding interaction, provide better thermal conductivity and more plasticity of the fuel.

3.2.3 Other materials

Carbon fibre reinforced silicon carbide (C/C-SiC) is a candidate material for the use in power plant applications, where extremely high temperature strength, thermal shock resistance, quasi-ductile material behaviour and good strength to weight ratio are advantageous. In inert atmospheres a good thermal stability is observed. The material could be used for gas turbine equipment (blades; vanes) but also for ultra high temperature heat exchangers.

The disadvantage is the insufficient gas tightness due to a inherent crack pattern caused by the manufacturing process and the limited oxidation stability. Multi-layer as well as gradated coatings are under development to cope with these weaknesses.

3.3 Improved Methods

3.3.1 Risk analysis

The probabilistic safety analysis (PSA) methodology has been well established as a tool for searching for improvements in the design of nuclear power plants. Utilising PSA techniques throughout the design process will result in optimisation of the plant safety balance. The optimisation also affects the plant economy by reducing the capital costs and increasing the plant availability. The use of PSA has been vastly affected by the development of computer efficiency. When the early PSAs were calculated in supercomputers the PSAs of today are calculated in workstations and PCs.

Present PSAs cover also exceptional situations such as low power operations, fires at the plant, extreme weather conditions and earthquakes. Today most plants have a so called living PSA which is used as an instrument also for the management of operational safety. Proposals have been made to bring the PSA as an instrument to the decision making in the control room and some such systems have been implemented in the world. Integrating passive safety systems into the well developed PSA methodology has been recognised as a future challenge.

3.3.2 Human reliability analysis

Human reliability analysis is an important part of the safety analysis. Present methods give some guidance on the search for possibilities of human intervention in various sequences of events. The difficulty however is to get believable reliability estimates. Various methods have been proposed for the assessment of various performance shaping factors by which reliability estimates have to transformed to reflect the environment where the interactions are made.

3.3.3 Organisational factors

Organisation factors have demonstrated their importance as contributors to human errors. The analysis of work processes can give an important contribution to better structures for authority and responsibility. The writing and maintaining of plant instructions and work procedures is another part which can be supported by information technology.

3.3.4 Decision analysis

Decision analysis gives a possibility to improve difficult decisions. In principle a decision consists of a set of decision alternatives, a set of utility functions and a systems model connecting decision alternatives to the utilities. Various software packages have been built to support the value elicitation process, data mining can be used to search for novel decision alternatives and various computer models can be used to predict the outcomes of selected decision alternatives. Sensitivity and uncertainty analysis can be used to indicate risks connected to a decision.

3.3.5 Software quality control

Verification and validation of software has become a major issue with the increasing complexity of software in general. The licensing of programmable automation for safety system applications is one very important area which has not yet found its satisfactory solution. The need to provide quantitative reliability estimates for a safety function may be difficult to meet. Various methods such as formal specifications, sneak-path analysis, software analysers, test environments, etc. have been proposed, but the provide only part solutions to the basic problem. For computer systems which are not safety critical standard software quality control systems often are enough.

3.3.6 Signal validation

Signal validation is today carried out only for the most important signals. In the future systems there are far better possibilities to do extensive validation. Model reference techniques can be used to compare signals from different sensors and react if anomalies are detected. Digital filtering and signal spectra can be used to detect possible problems in the believability of signals. Smart transmitters already contain extensive methods for alarming if a signal value does not appear to be valid. Smart transmitters typically do the linearisation and the conversion to engineering units to off-load other computers in the system.

3.3.7 Virtual reality

Virtual reality is another part of the simulation of the plant before it has been built. A virtual reality model of the plant can also be used as an input to advanced calculations connected to radiation doses, structural integrity, thermal stresses in piping, etc. Virtual reality can also be used for training purposes for example in maintenance activities.

3.3.8 New calculation methods

3D computational fluid dynamic (CFD) codes allow more realistic description of physical and chemical phenomena. The use of CFD codes focus on predicting local parameters (like concentrations, velocities, etc.). Due to the long computation times needed, CFD methods have been largely used as complementary to conventional analysis methods. The advancement of numerical methods and the increase of computer performane/cost ratio has made CFD codes more and more attractive choice of analysis tools. The advantages of the CFD codes include [NEA/CSNI/R(1999)7]:

- they can predict local effects that are important for accident consequences and mitigation strategies (hydrogen distribution, location of autocatalytic recombiners, operation of passive systems)
- they can simulate heat and mass transfer in a more realistic way,
- they enable description of coupled transport and combustion
- they can estimate counter-current flows in large flow paths,

Some of the disadvantages are:

- large effort needed for input
- long computation times
- limited possibility for parametric studies
- possibility for large user effects
- validation base is very limited

Application of the CFD codes is still developing. It needs further validation efforts and proof of predictive reliability. Detailed validation of the codes has suffered from restrictions due to measurement techniques suitable to provide detailed enough local data.

3.3.9 Narrow gap welding

Conventional V-shaped weld preparation of thick walled components requires a large volume of weld material, leads to high shrinkage of the weld. The high energy input results in unfavourable properties of the heat affected zone.

Narrow gap welding is aimed at optimising the integral behaviour of the weld joint and minimising the costs for welding. However, the technique requires very sensitive control of the process parameters and highly sophisticated equipment.

For economic application improved hardware and software has to be used. Important for the quality assurance process is the on-line evaluation of the temperature measurements in real-time with thermal video camera and/or laser systems which are implemented directly in the weld control process. The benefit of the narrow gap welding technique with on-line quality assurance and control systems is the high quality of the weld joint including low residual stresses and a more economic production process.

3.3.10 Shape welding

During the manufacture of thick-walled ferritic components, mainly pressure vessels and turbine shafts the problem exists, that nonhomogeneity and directionality in connection with segregations are decreasing strength and toughness of the material. Not infrequently embrittlement and/or cracking originates in the heat affected zone (HAZ) of weldments of such materials.

Multilayer shape welding technology allows the three-dimensional manufacture of large scale power plant components with a much lower input of material, compared to forging. The material behaviour of taylor-made products manufactured by this technology is almost isotropic and shows higher toughness and strength as compared to forged and rolled materials with the same content of alloying elements. Multilayer shape-welded materials are, due to their homogeneous structure, very well suited for ultrasonic testing.

It is important that mechanical properties for low and (very) high temperature application, corrosion resistance and residual stress distribution in thick-walled components can be adjusted and optimised by means of gradated material structure and/or heat control. A well assorted selection of additional weld materials enables the manufacture to produce steel components with a reduced concentration of accompanying elements like S, P and Cu. Detrimental trace elements can be limited to a nearly zero level. All theses advantages give also way for the manufacturing of smaller high quality products like elbows, nozzles and pressure tubes.

Future research work should concentrate on high strength and corrosion resistant bodies built up of gradated deposit materials for low and high temperature applications. Improved material models and thermo-mechanical simulation are necessary tools to predict local and global properties.

3.3.11 Electron beam welding

Electron beam welding is characterised by a well defined energy input and high penetration rate. Therefore the joining of thick-walled components of large dimensions is possible. The application can be extended to high strength and/or heat resistant steels. Compared to multi-layer weldments manufacturing and nondestructive testing is cheaper and safer.

One disadvantage, however, is that in the region of the heat affected zone (HAZ) micro structures with reduced toughness and high sensitivity to cracking may be produced. In case of circumferential welds, the quality of the weld in the closing area is also a problem.

Future research work should concentrate on innovative process control and quality assurance systems. Improved modelling and simulation of welding parameters and material conditions will help to optimise the dynamic process during welding and to control deformation and residual stresses.

3.3.12 Strain based design concept

The current pressure components in nuclear power plants have been designed on stress-based philosophy. This means, that maximum stresses must not exceed stress limits stated in codes. The limits encompass safety factors on the strenght of the applied materials. The basis of the failure limits is pure linear elastic behaviour of the material and 1 D behaviour. Real components, on the other hand, have changes of shape that cause the the load to be non-uniformly distributed, leadinh to stress and starin concentrations. Ductile materials, under continued load and beyond elasticity limit, allow redistribution of peak loads through local yielding, thus reducing the failure risk of the component.

The strain based design concept takes credit from the real load carrying capacity of the structures, hence reducing undue conservatism. The concept suggests a limit strain that should be used as a criterion for failure. The concept has been used in Germany for nuclear plant piping and for fast breeder reactor tank. The concept will be further studied in the EU 5th framework programme project LISSAC (Limits of Strain for Severe Accident Conditions).

3.3.13 Reactor pressure vessel annealing

The reactor pressure vessel base material is ferritic or perlitic steel, while the cladding is normally 316L or equivalent austenitic steel. Periodical Charpy impact testing of surveillance capsules allows evaluating the reduction in the upper shelf energy and the transition temperature shift which gives evidence of the increasingly brittle material behaviour.

The opening of the former Soviet Union countries to western countries inspections has allowed a deep exchange of results on the subject, and especially of mitigation procedures. CIS and CEEC installations have indeed already successfully experienced in-situ annealing procedures of the reactor pressure vessel, therefore demonstrating that the life of what is considered the unreplaceble NPP component can be practically extended.

The reason why east European NPPs needed annealing quite rapidly is the high content of phosphorus in the weld, As a consequence of irradiation embrittlement, phosphorus segregates at grain boundaries. Western base materials and welds contain on the other hand nickel, which also makes the steel sensitive to neutron irradiation.

New studies are therefore necessary to assess the influence of chemical composition (mainly nickel, copper and phosphorus) on the impact curve transition temperature shift of RPV ferritic steels.

Several questions are moreover still opened on the subject of annealing: with this treatment a steel can recover up to 90% of its fracture toughness, in terms of transition temperature and upper shelf Charpy impact energy, but the mechanism by which this happens is not yet clear, because phosphorus segregation at grain boundaries is not eliminated by the heat treatment and therefore other factors must play a role.

Subject of further studies is moreover the behaviour of the annealed steel undergoing re-embrittlement. The extent to which the transition temperature shift increases can be foreseen according to different hypotheses which need to be verified.

Another aspect that needs to be further investigated and is planned to be studied is the influence of fluence rate on irradiation embrittlement. There are indeed contrasting results showing direct correlation in some cases, while in others no significant dependance has been found even if the fluence rate was increased up to five orders of magnitude higher.

Once again it's the chemical composition that plays the determinant role in affecting the sensitivity of the material.

4. R&D Proposals and availability of necessary R&D facilities

The SESAR/FAP group working within OECD/NEA/CSNI has produced a comprehensive list of material testing facilities [SESAR/FAP 1999]. The facilities identified are collected in the following tables 4-8.

Table 4. Material Test Reactors

Facility	***Specifications***
NRU (Canada	Decommissioned after 2001
OSIRIS (France)	
FMR (USA)	
BR2 (Belgium)	
Halden (Norway)	
HFR (The Netherlands)	high availability: 0.8 y (285 d/y) operational specialisation: >50%, irradiation of structural mat.
LVR-15 (Czech Republic)	

Table 5. Hot Cells Facilities: mechanical materials testing, microstructure characterisation, analytical tools for PIE

Facility	***Specification***
ANL (USA)	mechanical testing, microstructural characterisation, corrosion
ORNL (USA)	mechanical testing
AEA T (GB)	mechanical testing, microstructural characterisation, corrosion
Magnox, Berkeley (GB)	mechanical testing, microstructural characterisation
CEA (France)	mechanical testing, microstructural characterisation, corrosion
ITU Karlsruhe (Germany)	mechanical testing
JAERI (Japan)	mechanical testing, microstructural characterisation, corrosion
NRG (The Netherlands)	series mechanical testing cells (11), specialised facilities for fatigue and fracture mechanics testing (class A)
NRI (Czech Republic)	mechanical testing, microstructural characterisation, corrosion
PSI (Switzerland)	mechanical testing, microstructural characterisation, corrosion
CIEMAT (Spain)	mechanical testing
SCK/CEN (Belgium)	mechanical testing

Table 6. Large Scale Structural Testing Facilities

Facility	***Specification***
AEA T (GB)	100MN biaxial test rig, spinning cylinder test rig
TWI (GB)	medium-scale testing for fatigue
ORNL (USA)	biaxial test capability
Battelle (USA)	large test facility for piping, large test facilities for fatigue and fracture
NIST (USA)	large test facility for fracture, medium test facility for fatigue and fracture
MPA Stuttgart (Germany)	100MN Universal test rig, explosion driven fracture. test facility
CEA (France	medium-scale testing for fatigue
ISMES Bergamo (Italy)	large test facility for components
JRC Ispra (Italy)	large Hopkinson impact facility
PSI (Switzerland)	large test facility for components, biaxial test capability
NRI (Czech Republic)	80 MN Universal test rig
Skoda JS Plzen (Czech Rep.)	10 MN low-cycle machine
Prometey (Russia)	large test facility for components

Table 7. Non-Destructive Examination (NDE), especiallly large components with flaws

Facility	***Specification***
AEA T (GB)	RPV test sections
NRI (Czech Republic)	VVER RPV test sections
ANL (USA)	steam generator tubes
Pacific Northwest NL (USA)	austenitic and dissimilar welds
EPRI (USA)	RPV test sections, austenitic and dissimilar welds
JRC Petten (The Netherlands)	PISC test pieces
MPA Stuttgart (Germany)	RPV test vessel
IfzP Saarbrücken (Germany)	austenitic and dissimilar welds
Prometey (Russia)	VVER RPV test sections
Skoda (Czech Rep.)	VVER RPV test sections

Table 8. Environmentally assisted cracking (EAC) laboratories, including corrosion autoclave facilities

Facility	***Specification***
MHI Takasago (Japan)	numerous EAC rigs, various large SC test facilities
Studsvik (Sweden)	rig in materials test reactor
ABB-Atom (Sweden)	
AEA T (GB)	
ANL (USA	numerous EAC rigs
EPRI (USA)	numerous EAC rigs
GE (USA)	numerous EAC rigs
VTT , Espoo (FIN)	EAC, IASCC rigs
Siemens KWU (Germany)	various autoclaves and loops
MPA Stuttgart (Germany)	various autoclaves and loops
JRC Petten (The Netherlands)	rig in materials test reactor
CIEMAT (Spain)	10 loops with autoclaves, 6 static autoclaves
NRI (Czech Republic)	2PWR+2BWR SC loops in LVR-15 test reactor, various autoclaves and SC test facilities
PSI (Switzerland)	2BWR SC loops with static/cyclic loading

5. Recommendation on Future R&D Activities

The fundamental requirement for safety critical I&C systems is very high reliability. For digital I&C systems it has turned out very difficult to formally prove the reliability required, which has resulted licensing problems. Research is needed to improve the verification and validation techniques needed to demonstrate that the reliability requirements are met. Most of the faults in modern high-integrity systems originate from errors in the initial specification. Formal techniques should be developed for specification of systems agaist their safety requirements. They should be supported by analysis tools to permit effective verification. There is a need to further develop the man-machine interface for new type of control rooms. The use of computer based operator aids is an evolving area. A closely related are is instrument survival in accident conditions (especially in severe accidents) and signal validation techniques.

Energy production processes at high temperature can dramatically increase the efficiency in electricity production and thus reduce the necessary amount of fuel and radioactive waste. Existing ODS alloys have to be improved with regard to their anisotropic strength behaviour. The long term behaviour under service like loading situations (environment, mutliaxial creep-fatigue loading) has to be investigated from which damage hypotheses and constitutive material laws can be derived to be implemented in life time assessments. For reinforced ceramics suitable coatings must be developed. The effectiveness of the coatings under cyclic and thermal shock loading has to be developed with regard to micro-cracking, peeling-off and oxidation protection.

PSA techniques are already widely used and they have proved to useful to achieve balanced plant design. PSA should be a routine tool in design and licensing of future nuclear plants. The new challenge for PSA methods is to integrate the passive systems into the same framework. Common mode failures and maintenance considerations should be improved. CFD codes are used already now for specific solutions. Fos safety eveluations, methods and data for CFD code validation shoud be established.

New welding techniques allow lower cost and joining of large components. The inspection possibilities will be improved. Future research work should concentrate on innovative process control and quality assurance systems. Improved modelling and simulation of welding parameters and material conditions will help to optimise the dynamic process during welding and to control deformation and residual stresses.

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