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Note on the Outcome of the June 1989
CSNI Specialist Meeting on

# INTENTIONAL COOLANT SYSTEM DEPRESSURISATION

Prepared by
the Senior Group of Experts on Severe Accidents

**COMMITTEE ON THE SAFETY OF NUCLEAR INSTALLATIONS**
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# OECD

In 1948, the United States offered Marshall Plan aid to Europe, provided the war-torn European countries worked together for their own recovery. This they did in the Organisation for European Economic Co-operation (OEEC).

In 1960, Europe's fortunes had been restored; her standard of living was higher than ever before. On both sides of the Atlantic the interdependence of the industralised countries of the Western World was now widely recognised. Canada and the United States joined the European countries of the OEEC to create a new organisation, the Organisation for Economic Co-operation and Development. The Convention establishing the OECD was signed in Paris on 14th December 1960.

Pursuant to article 1 of the Convention, which came into force on 30th September 1961, the Organisation for Economic Co-operation and development 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 this 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 Signatories of the Convention were Austria, Belgium, Canada, Denmark, France, the Federal Republic of Germany, Greece, Iceland, Ireland, Italy, Luxembourg, the Netherlands, Norway, Portugal, Spain, Sweden, Switzerland, Turkey, the Unided Kingdom and the United States. The following countries acceded subsequently to the Convention (the dates are those on which the instruments of accession were deposited): Japan (28th April 1964), Finland (28th January 1969), Australia (7th June 1971) and New Zealand (29th May 1973).

The Socialist Federal Republic of Yugoslavia takes part in certain work of the OECD (agreement of 28th October 1961).

# NEA

The OECD Nuclear Energy Agency (NEA) was established on 20th April 1972, replacing OECD's European Nuclear Energy Agency (ENEA, established on 20th December 1957) on the adhesion of Japan as a full member.

NEA now groups all the European Member countries of OECD and Australia, Canada, Japan and the United States. The Commission of the European Communities takes part in the work of the Agency.

The primary objectives of NEA are to promote co-operation between its Member governments on the safety and regulatory aspects of nuclear development, and on assessing the future role of nuclear energy as a contributor to economic progress.

This is achieved by:

-- encouraging harmonisation of governments' regulatory policies and practices in the nuclear field, 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;

-- keeping under review the technical and economic characteristics of nuclear power growth and of the nuclear fuel cycle, and assessing demand and supply for the different phases of the nuclear power to overall energy demand;

-- developing exchanges of scientific and technical information on nuclear energy, particulary through participation in common services;

-- setting up international research and development programmes and undertakings jointly organised and operated by OECD countries.

In these and related tasks, 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.

# CSNI

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 coordinate 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 on technical issues of common interest. It promotes the coordination of work in different Member countries including the establishment of co-operative research 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.

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# FOREWORD

The primary objective of operator actions during an accident involving loss of coolant or a transient is to bring the plant to safe and stable conditions after shutdown of the reactor. To achieve this goal, in some of the design basis events, for instance a steam generator tube rupture in a PWR, operators may be required to depressurise the reactor; in BWRs, the system pressure is reduced automatically through an automatic depressurisation system.

Primary coolant system depressurisation is already taken into account in accident procedures designed to cope with total loss of feedwater by establishing a feed-and-bleed configuration. In severe accident analysis, it has been suggested that intentional reactor vessel depressurisation before melt-through could stop a high pressure accident sequence and associated high pressure melt ejection, and that it should be considered in accident management strategies.

It has also been suggested that in certain accident situations depressurisation could provide possibilities for core cooling with low-head systems in order to avoid or limit fuel damage. Also in this case is the secondary system in PWRs to be taken into account. Low circuit pressure generally increases the number of options allowing to control the situation; it also makes possible the use of additional equipment, such as mobile pumps.

However, uncertainties remain regarding the effectiveness of depressurisation accident management strategies and possible negative effects in situations exceeding the design basis. Any severe accident would probably be the consequence of a number of failures and mistakes resulting in an unclear situation for the operators. It is under such difficult circumstances that a decision to depressurise would have to be taken. General guidance for the operators could be given in procedures but the progression of a real accident situation will always be specific. Reducing pressure through blowdown implies loss of coolant; in some situations this could mean early start of fuel heatup and failure, and the use of equipment under conditions exceeding the design basis.

At its sixteenth meeting held in November 1988 NEA's Committee on the Safety of Nuclear Installations (CSNI) agreed that a meeting should be organised to provide specialists from licensing bodies, research organisations and industry with an opportunity to present and discuss the progress of work underway, foreseen applications of intentional coolant system depressurisation, and the associated specific problems. The Gesellschaft für Reaktorsicherheit (GRS) mbH kindly offered to host the meeting in Garching, Federal Republic of Germany, from 12th to 14th June 1989.

The meeting attempted to give answers to questions in the following areas:

1) Basic objectives of intentional coolant system depressurisation

2) Design objectives and instrumentation

3) Existing or planned safety studies and assessment; phenomenology of coolant system depressurisation

4) Management strategies for intentional coolant system depressurisation

5) Actual designs, operational experience, availability, testing and qualification

6) Identification of needs for additional studies

The Specialist Meeting on Intentional Coolant System Depressurisation was organised by a Programme Committee composed of Members of CSNI's Senior Group of Experts on Severe Accidents or their representatives:
Prof. Dr. E. F. Hicken (Chairman of the Programme Committee and of the Specialist Meeting), Mr. P. Bystedt, Mr. Y. Cornille, Dr. M. Réocreux, Dr. K. Soda and Dr. T.P. Speis. It was followed by a meeting of the full Senior Group of Experts on Severe Accidents which discussed the results of the exchange of information and prepared summary conclusions and recommendations for CSNI; the outcome of this meeting is presented in the following document.

The full Proceedings of the Specialist Meeting, prepared by GRS, have been published as CSNI Report 158.

The Secretariat

# INTENTIONAL COOLANT SYSTEM DEPRESSURISATION

## A. STATEMENT OF THE ISSUE

One of the measures open to consideration in the course of a severe accident to an LWR is deliberate depressurisation. Even within the design basis it may be contemplated as a response to Steam Generator Tube Rupture (SGTR) since for such an event there are clear advantages in reducing the pressure differential between the primary and secondary side. In BWRs, Automatic Depressurisation Systems (ADSs) are standard provision. For PWRs the secondary, as well as the primary, may be pressurised, both actions achieving a reduction of primary circuit pressure. This brief report sets out to review more generally the potential advantages of depressurisation; what the disadvantages may be; the requirements placed on operators and equipment; and the prerequisites for successful implementation.

## B. WHY DEPRESSURISE INTENTIONALLY?

The essential reason is to maximise flexibility in countering accidents. At an early stage of accident development the range of available water supplies is increased by lowering primary circuit pressure because of the greater ease of injecting water. Consequently, the operator's ability to reduce the probability that a serious accident progresses to a full core melt is enhanced. Even if depressurisation does not result in restoring or maintaining adequate cooling to the core, and the accident progresses to an advanced stage, there are circumstances in which the extreme consequences of core melting may be significantly mitigated by depressurising. At a lower pressure the stress in the reactor pressure vessel wall and the driving force for rapid and wide dispersal of finely fragmented fuel will be reduced.

Nevertheless, there are other aspects which are less favourable and which need to be considered. First, the loss of coolant from the primary circuit is usually undesirable in principle and therefore the operator would be required to act counter to his normal training and inclinations. The perceived benefits may not be achieved in all situations, and the advantages need to be studied carefully over a range of reactor states and postulated accidents. It has been suggested that in severely degraded core conditions a lower pressure may facilitate triggering of a violent thermal interaction between molten fuel and water. Moreover, the provision of equipment and procedures for deliberate depressurisation carries the risks of equipment malfunction and of operator error by initiating or continuing depressurisation in inappropriate circumstances.

Whilst the Senior Group believes that in most instances it will be beneficial to provide operators with the option of deliberately depressurising the primary circuit, there are a number of provisos to be stressed. They include training of personnel, provision of the best practicable diagnostics, the careful and thorough development of validated procedures, and the installation of equipment properly qualified for the conditions in which it is required to operate. Above all, the ability must be maintained to halt the process of depressurisation if changing conditions (eg restoration of electric power) demand it.

These questions are addressed in more detail in the later sections of this report. Relevant phenomena are reviewed in some detail, followed by a discussion of accident management principles in relation to intentional depressurisation. Attention is drawn to the need to understand and provide for the unusual conditions in which key components, particularly valves, would be required to operate when implementing accident management measures beyond the design basis.

## C. DEPRESSURISATION PHENOMENA

### INTRODUCTION

Intentional depressurisation of the RCS offers a priori significant advantages. In addition to enabling low pressure injection systems to be used, which constitutes an important contribution to the core melt prevention, depressurisation of the RCS in the course of a severe accident is an obvious way to avoid high pressure ejection of molten core debris to the containment and so mitigate the resulting problems of DCH and vessel-lift by reaction forces. Although the depressurisation may influence various phenomena on which there are large uncertainties, such as the possible increase of the risk of energetic interaction between water and corium at low pressure, there is no doubt of the interest of the depressurisation as a management strategy considering that in practically all situations for which this measure would be considered, the natural course of the accident would lead to the rupture of the vessel and the depressurisation of the RCS in worse conditions. Indeed the attraction and the viability of this measure depend very much on plant characteristics such as reliability and discharge capacity of valves, design of the vessel and of the vessel pit. Clearly therefore for plant which may need significant adaptation to achieve efficient depressurisation that question and issues related to the consequences of the depressurisation should be resolved before making the decision to depressurise as an accident management strategy. In the sections that follow the present state of knowledge and understanding of phenomena which are relevant to the decision-making process will be discussed.

### DETERMINATION OF DEPRESSURISATION RATE/LIMITING CONDITIONS AFFECTING FINAL PRESSURE

Primary feed and bleed have the capability to remove decay heat and reduce RCS pressure provided that:

(a) the energy flow rate out of the power/pilot operated relief valves (PORVs) is greater than decay heat; and

(b) the in-flow from safety injection is sufficient to maintain the RCS coolant inventory.

The depressurisation rate is primarily a function of the total PORV flow area and pressure for a pure steam flow.

The depressurisation rate also is a function of the steam quality upstream of the PORVs. For a given area and upstream pressure, the depressurisation rate will be a maximum for a pure steam flow since the maximum energy is removed. The rate will be lower for two-phase conditions. The steam quality upstream of the PORVs is a function of the quantity of liquid entrained as droplets in the steam flow (assuming that the interface level is below the top of the pressuriser) as steam bubbles in the liquid ascend and break the surface. If the PORV steam flow rate is sufficient to reduce RCS pressure to the accumulator set point, then the accumulator flow will be initiated. Pressure is then controlled by the accumulator injection process for a long time (hours). After the accumulators are depleted, the pressure falls further since more system inventory is lost through latched open PORVs.

Secondary side depressurisation is also an option for providing cooling to the primary system. The rate of heat transfer will depend on secondary side saturation temperature, which in turn will depend upon the rate at which the secondary side can be vented.

## ANALYSIS METHODS

Depressurisation calculations carried out to the point at which core damage starts are usually performed with standard transient analysis codes used to calculate small break accidents. Codes such as RELAP (US), ATHLET (FRG) and CATHARE (France) are typical of the codes used. Verification of these codes up until the point of core damage is fairly extensive owing to their use in design basis accidents and hence high confidence can be given to calculated results (eg depressurisation rate). Once core damage is predicted to begin, different codes must be used and the uncertainty in the calculation goes up due to lack of extensive verification data. Nevertheless, recent calculations using MAAP (W Brussels), Thales-PM (Japan) and RELAP/SCDAP (US) all showed reasonably good agreement regarding the general conclusions that can be drawn from depressurisation studies.

## EFFECTS OF WATER INJECTION

There remain some phenomena associated with depressurisation for which a good understanding still does not exist. These phenomena centre around: reflood of a hot/damaged core, condensation heat transfer between injection water and the steam/hydrogen mixture, pressuriser water inventory and its impact on the system pressure, steam generation and associated pressure spike produced due to interactions of molten core materials and water, failure (location and size) of the RCS piping, and debris formation. Experimental and analytical programmes in several countries are addressing these phenomena (eg fuel-coolant studies in the UK, the US, France and FRG).

## EFFECTS OF DEPRESSURISATION ON EQUIPMENT AND SYSTEMS

Once significant oxidation of fuel cladding has begun, depressurisation through the PORV will move high temperature steam and non-condensable gases from the core into the hot leg and pressuriser surge line. Transport of high temperature gas within the surge line could lead to creep rupture or melting. The estimates of heat transfer to a component, such as the pressuriser surge line, are based on single phase heat transfer correlations that are well known, widely used and with known uncertainties (± 20%). Creep rupture and melting properties for steel are well known and based on simple laboratory tests. The major uncertainty is in the energy transport process. A determination of the flow paths is required to determine where the energy is deposited in RCS structures.

The pressure loads during early (before significant cladding oxidation) depressurisation are not expected to cause any concerns in RCS piping other than perhaps to steam generator tubes. The pressure loads on steam generator tubes during depressurisation need to be examined if the secondary side is maintained at pressure.

Normally the PORVs and safety relief valves discharge into a drain or pressure relief tank. These tanks are protected against catastrophic failure by a rupture disc. This is a normal overpressure safety function, and should prevent missile hazards during the blow-down. If there is significant oxidation of the fuel cladding, the discharge temperatures may be high enough (if the pressuriser liquid level is zero) to result in auto-ignition of any discharged hydrogen. Containment temperatures alone may be high enough to cause equipment damage. Combustion or detonation of hydrogen may add to the potential for equipment damage.

Programmes within the FRG and the US have been actively investigating hydrogen burning in containments. Experiments on flame propagation and stability, deflagration, deflagration to detonation transition, and detonation have been conducted. Model containments in the FRG have been used to study hydrogen stratification and mixing in containment. This work has rendered a very clear picture of hydrogen burning in containments. However, the behaviour of high temperature hydrogen in a high steam environment is still unknown.

## PRESSURE EFFECTS ON MELT EJECTION

Successful depressurisation of the reactor coolant system to avoid DCH can be realised if:

(a) the residual energy in the primary system at the time of vessel failure is insufficient to eject a substantial quantity of the molten core material out of the reactor cavity, or

(b) the combined containment load due to transferring the core debris thermal energy to the containment atmosphere, the chemical interactions with the steam and air in the containment, and the combustion of the generated hydrogen is below the containment failure pressure.

Studies in the areas of HPME and DCH generally assume the primary system pressure to be above the accumulator set point. At this pressure, the results of experiments conducted in the US obtained from using several cavity models (without any imposing structures such as steam generator compartments) suggest that nearly complete debris dispersal is expected. Scaled experiments conducted within the UK indicated that in-cavity structures beneath the reactor vessel may appreciably retard the dispersal of ejected core debris.

NRC DCH research is addressing the question of at what reactor coolant system pressure will DCH induced containment loading be sufficiently low as not to challenge the integrity of the containment. This reactor system pressure is termed the low pressure cut-off. Experimental observations from the low pressure experiments indicate that:

(a) Under a variety of conditions tested, the initial pressure required for the complete dispersal of melt simulant from the cavity was nearly the same and was above the accumulator set point.

(b) The presence of structures in the cavity has the influence of retaining more melt at higher pressures but does not significantly affect the initial pressure required for complete dispersal.

(c) The initial mass of melt simulant present does not affect the initial pressure required for complete dispersal, but results suggest that the entrainment rate may be proportional to the amount of melt simulant present.

(d) Dispersal increases with increased RPV breach area.

(e) The low pressure cut-off correlations developed suggest that a different correlation is needed for each cavity design studied.

Final conclusions as to the utility of low pressure cut-off correlations for full scale melt dispersal predictions are not yet known. Melt dispersal experiments to date have not simulated all the relevant cavity phenomena such as steam-melt chemical reactions, dissolved hydrogen or gas blow-through during melt discharge phase.

## REACTION FORCES ON EQUIPMENT

It has been postulated that failure of an LWR reactor vessel from molten core debris falling on to the bottom head while the

RCS was at high pressure might result in significant upward thrust on the reactor vessel, perhaps sufficient to lift the vessel so that it can impact on the containment and cause a breach. In addition the FRG has been particularly concerned with this possibility in PWR designs in which the cavities beneath the reactor vessel are closed, adding the force of expanding gases to the upward thrust of the reaction force. Calculations performed analysing this possibility indicate that for US type PWRs, under worst case assumptions, the force will not be sufficient to break loose the reactor vessel from the RCS piping and that the vessel will be arrested after about 1 ft of travel. However, this may not be the case for the closed cavity design.

## NATURAL CIRCULATION EFFECTS DURING DEPRESSURISATION

Experiments performed at the Westinghouse 1/7 scale facility suggest that opening the PORV to depressurise the primary system will inhibit natural circulation in the RCS. Calculations of RCS natural circulation also suggest this. However, if the PORV is closed, or never opened initially, both experiments and calculations suggest that natural circulation will be established quickly in the RCS. Once significant cladding oxidation has begun, the circulation of hot steam and non-condensable gases in the RCS will quickly heat the hot leg and steam generator. Calculations indicate that the pressuriser surge line is likely to fail by creep rupture and depressurise the RCS prior to the onset of fuel melting. Clearly, there are other possibilities, including failure of a steam generator tube which will lead to containment by-pass, but at this time surge line failure seems to be the most likely prospect. [Related calculations suggest that if the PORV is latched open, the hot gases exiting the pressuriser also will fail the surge line.] If these calculations are correct, the subsequent failure of the reactor vessel will be at low RCS pressure, precluding DCH. However, the natural circulation calculations that were performed were specific to a Westinghouse PWR (Surry) and their overall validity and the general applicability of the conclusions drawn have not yet been tested.

## LOADS FROM FUEL-COOLANT INTERACTIONS

Fuel-coolant interaction is a topic of considerable interest, particularly with respect to accident management. Adding water to a degraded core, or molten debris slumping into an LWR lower plenum, will generate large amounts of steam and hydrogen. Further, the interaction itself could be quite energetic. Experiments to date seem to suggest that "steam explosions" are more likely to be "triggered"[1] at lower pressures; on the other hand, the probability of assembling larger pre-mixtures is larger at higher pressures. It is the US

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[1] Triggering is the term used to denote the phenomena characterised by the collapse of a vapour film surrounding a hot melt particle, resulting in high heat transfer rates, large relative fluid velocities near the melt particle, and generation and propagation of a shockwave. The phenomenon is typically characterised by a timescale of about 100 microseconds.

current judgement that the previously subjectively estimated upper bound value for the ex-failure probability of 0.01 (NUREG-1116, "A Review of the Current Understanding of the Potential for Containment Failure From an In-Vessel Steam Explosion") will not be substantially affected by these two opposing effects[2]. A number of programmes are under way to gain more insight into these effects and further enhance understanding of the steam explosion issue.

For example, some programmes are under way to investigate the rate of fragmentation under propagating explosion conditions. Pre-mixing experiments are also being conducted to test the predictive capability of transient multi-field models developed to analyse steam explosion phenomena.

There is a fair amount of data on mixing in pouring modes at medium scale, although much of this data is not well characterised and is not very suitable for model assessment.

With regard to ex-vessel loads, some initial calculations were done for the Severe Accident Risk Study (NUREG-1150) using CSQ-II to calculate loads from an assumed explosion source. These "explosions" were postulated additions of water to molten debris that had melted through the reactor vessel and was lying in the pedestal area beneath the vessel. The loads calculated were much less than those derived earlier from simplified calculations that did not take account of possible pressure relief paths.

## D. ACCIDENT MANAGEMENT

### GENERAL

Accident management (AM) as a method and guide for assuring safety functions in accident situations has in recent years become more and more important. AM is important for accident situations covered by design basis events, but most of all in situations when the emergency safety functions are more severely degraded. The general objective for AM is to bring the plant to a final stable state after an accident, with a coolant system without substantial overpressure and with secure confinement of radioactive material.

Generally, the procedures for AM now in use or under development are symptom- or performance-oriented and based on realistic evaluation of plant behaviour. With properly implemented procedures for AM, the responsible plant personnel are in a better position to avoid core damage by the initiation of preventive measures and, if core damage does occur, to limit plant damage and the radiological consequences by the initiation of mitigative measures.

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2. NUREG-1150 (Appendix C - Vol. 2) "Severe Accident Risks; An Assessment for Five US Nuclear Power Plants" - Second Draft for Peer Review - June 1989.

A typical situation when the objective for AM is preventive would be when high pressure make-up capacity is insufficient to maintain cooling but low pressure capacity is operable. For PWRs, this could be the situation for the secondary side as well as for the primary side. In both cases preventive action could take the form of reducing system pressure sufficiently to enable low pressure systems to come into play. Another preventive objective of intentional depressurisation is, in the case of a steam generator tube break in a PWR, to minimise the loss of primary coolant into the secondary side and also to prevent overfilling of the defective steam generator.

In situations with more severe equipment failure, mitigative objectives could be to gain time for the attempts to restore power or alternative cooling modes before core melt. For example, for loss of electric power, depressurisation of a PWR primary system to a pressure level where accumulators could discharge coolant into the system could temporarily restore cooling and delay the onset of core heat-up. At a later stage in the sequence a mitigative objective could be reducing the pressure in the reactor vessel before a vessel failure as a result of core melt takes place. Such measures would address the concern aroused by the phenomenological uncertainties associated with high pressure melt ejection and the potential for early containment failure caused by direct containment heating. In a containment by-pass event, such as a leak in a non-isolated system, depressurisation of the reactor system would reduce the loss of coolant and the possible release of radioactive material.

Intentional depressurisation of coolant systems is one part of the AM measures. Its possible use by operators will vary greatly depending on the type and specific features of the plant, such as instrumentation, valve capacities, cooling options, etc., and on the specific situation at hand. Whilst some such situations would challenge the operators in terms of decision making and performance, AM measures may be expected to minimise the challenge.

## MEANS FOR INTENTIONAL DEPRESSURISATION

Nuclear reactors are equipped with several means for depressurisation that could be utilised by the operators in AM. Typically, PWRs have systems for pressuriser spray, for steam dump to the turbine condenser or to the atmosphere, and for coolant discharge from the primary system. The preferred way of depressurisation is to cool down the primary side through bleed and feed from the secondary side. In that way, primary side water is conserved and the secondary side becomes more accessible for cooling in alternative modes. Only if this is not feasible, because of loss of system performance, should the discharge of primary coolant be used. Some PWRs have special capabilities to feed the secondary side at low pressure if main and auxiliary feed systems are unavailable. Examples are: injection from the feed water tank and back-up from other sources such as fire water systems.

BWRs have relief valves on the steam lines. Automatic depressurisation through these valves will be initiated by the protection system on certain signals. The valves can also be opened manually. For limited capacity back-up, drainage systems from the steam lines can be used at some plants. Some of the older BWRs have isolation condensers that could be used for depressurisation without loss of coolant from the reactor system.

## INITIATION

The general role of the operators in an accident situation is to identify the plant status, to verify proper automatic system function, and to take actions in order to mitigate damage to the plant and to mitigate radiological consequences. If system malfunctions occur, the operators should restore proper functions or find alternative ways to manage the situation.

The initiation of depressurisation will depend on the evaluation made by the operators of the specific situation they have at hand. However, they must be guided by instructions based on analysis of system response so that timing is adequate to meet the objectives. To be preventive, depressurisation must be initiated early enough for fuel damage by overheating to be avoided at least until low-pressure cooling capacity is sufficient. To be mitigative, eg in order to avoid high pressure melt-through, initiation must be timely in relation to relief valve capacity and equipment qualification for the specific mode of operation and environmental conditions. Generally, in situations when intentional depressurisation is to be used, an unusual event is under way and water losses from cooling systems may have occurred.

For an event in a PWR with loss of feed water at high pressure but with possibilities for low pressure feeding, secondary side depressurisation can be initiated when attempts to restart high pressure feed have been unsuccessful and low pressure alternatives are available. The rate of depressurisation should be chosen, taking into account necessary heat sink requirements and the stress limits during cool down, and should be included in the AM procedures.

If no feed to the secondary side is available or in prospect on the required timescale, the secondary heat sink will be inadequate to prevent heat-up and pressure increase in the primary system. Extraction of decay heat must then be done through bleed via the pressuriser relief valves. Initiation of bleed must be governed by plant characteristics, such as head capacity of high pressure injection systems, and the degradation of different systems in the specific situation. Typical symptoms for bleed initiation could be low level in the steam generators with no available feed capacity, or low level and high primary pressure.

If total loss of feed capacity has occurred for the secondary as well as for the primary side, the important parameters for initiation of depressurisation of the primary system are core uncovery and start of core heat-up. The objectives for depressurisation must then be to maintain adequate core cooling by use of primary inventory until accumulator discharge is initiated. Criteria for initiation (which for some PWRs include core exit temperatures in the range of 600-700°C) must be established on plant specific parameters and valve capacities. If significant core heat-up occurred, steam and hydrogen generation during pressure relief could keep the pressure up and prevent or delay the accumulators from discharging.

To avoid a high pressure melt-through scenario, it is necessary to optimise time of initiation and, again, analysis is necessary on the plant specific parameters.

For a steam generator tube rupture event (SGTR), the prime objective for primary side depressurisation is to limit the leak into the secondary side and to avoid overfilling of the damaged steam generator. When pressures have equalised, cool down is made to RHR system pressure. Depressurisation via steam dump from intact steam generators can be initiated as soon as the SGTR event has been identified. The cool down rate must be balanced to maintain adequate core cooling, increasing the rate if necessary on indications of increasing core temperature.

In a situation with a small break LOCA, the objectives would be similar: to achieve rapid reductions in break flow and primary system pressure. Depending on the break size and on available safety injection capacity, secondary and/or primary side relief can be chosen, though, as stated earlier, secondary depressurisation is to be preferred whenever possible. If, in the particular circumstances beyond-design basis conditions, injection capacity is insufficient and the core temperature starts to increase, depressurisation can be increased so that accumulator discharge can be obtained.

For BWRs one can see the following principal situations. If, because of multiple failures, high pressure feed capacity is insufficient to maintain the water level in the vessel, boil-off will lead to the start of core heat-up when the core becomes uncovered. A steam blow-down from the vessel that can remove energy in excess of decay heat will enhance coolant circulation in the vessel, increase the two-phase coolant level, and keep fuel temperature down by steam flow cooling of exposed parts. Thus, fuel temperature increase can be limited during blow-down to reflood or spray the core. Automatic depressurisation is initiated on certain low level signals, in some BWRs in combination with containment signals.

Manual depressurisation by the operators should be the first back-up to automatic initiation. If automatic initiation fails, the operators can initiate a proper blow-down by single valve operation to meet the objectives of depressurisation. However, they should first confirm that criteria for depressurisation have been met and that the automatic depressurisation system has actually failed.

Bringing the pressure in the reactor vessel down will open up possibilities for the use of various cooling systems, such as low head safety systems, feed water tanks (if installed) or condensate pumps (if available), or even make-up water or fire water (where connection to such a system is installed).

If the core cooling is unsuccessful and the core starts to degrade, it is important that the pressure relief capacity is maintained. Even if capacity needs are relatively low during heat up of the core, the demands can increase later due to contact between water and corium and heated vessel internals, and thus increase pressure at vessel failure.

## INSTRUMENTATION AND COMPUTERISED OPERATORS AID

Appropriate instrumentation is necessary to identify the plant status, to allow the plant personnel to initiate appropriate measures, and to follow the progress of the transient. The instrumentation can be divided into that necessary before core damage and that capable of measuring variables and able to withstand loads during severe core damage states.

In the first stage of the transients, before core melting has occurred, existing instrumentation will generally be sufficient. However, this has to be proven for each specific plant. Some additional instrumentation might be necessary to reliably initiate the depressurisation. The installation of water level measurement in PWR vessels or of core exit temperature measurement should be studied as well for these beyond-design situations.

For sequences with core degradation, only in a few countries are the reactors equipped with extended instrumentation or at least instrumentation with an extended range. Loads during those sequences (eg elevated temperatures in valves or connecting pipes or temperature and pressure loads from a hydrogen burn) have to be included in the specifications for related instrumentation. The operator and other responsible plant personnel should be equipped to diagnose the plant state and to predict further developments (the analysis of the sequence history is of less importance during the transient). Although diagnostic systems to assist the plant personnel are under development, more attention should be given to methods that allow the prediction of the further progress of the transient.

## QUALIFICATION OF VALVES

The main components utilised to depressurise a system are valves, i.e. for PWRs the valves on the secondary side of steam generators and on the pressuriser, and for BWRs on the steam lines. Those valves may be located inside or outside the containment.

These valves must accommodate saturated steam flow and sometimes sub-cooled or two-phase flows before core degradation.

During the core degradation process, valves connected to the primary coolant system may be exposed to high temperature non-condensible gas or steam flows.

Valves used for design basis sequences are qualified for related loads. These valves are also qualified for beyond-design sequences as long as the loads are within design limits, which is the case for most sequences with intentional depressurisation. However, for some sequences the loads on valves, including external loads from the containment, are beyond those for design. Experience and engineering judgement lead to the assessment that the valves would operate under the more severe conditions, at least for some time. Valves and related equipment have to be checked - and, if necessary, to be improved - for loads resulting from depressurisation. In addition, the realistic mass flows under these circumstances have to be evaluated.

## HUMAN FACTORS

Many of the actions within AM associated with depressurisation would be carried out in the time frame of 0.5 to 2 hours after the onset of the accident. It is therefore essential that the actions are governed by appropriate procedures. These must be developed on the basis of careful and realistic analysis of possible events, taking into account the specific characteristics of the plant and also the ability of the responsible plant personnel to identify the situation and act properly. Because every possible event cannot be foreseen, the procedures must also admit flexibility.

Intentional depressurisation should not result in a worse situation than if no action was taken. This must be taken into account in the analysis and in the procedures. The status of the plant, as evaluated by the personnel, is likely to change during the event. It must therefore also be possible for the personnel to interrupt depressurisation, once initiated, and to manage the situation from there.

In certain situations, intentional depressurisation may not be the natural response of the personnel, and they would rather rely on the possibility of restoring cooling so as to turn the event around before core degradation starts. Depressurisation must therefore be preceded by a firm and deliberate decision to choose a strategy which could have such great influence on plant shut-down and recovery. One can also see situations where there is a race between executing such a decision and the attempts to locate and restore failed systems in the plant (eg power supply or cooling systems). For such events, technical as well as management support to the control room crew will be essential. It is also essential that responsibilities and organisation are clearly defined both within the operating crew and between them and the support personnel. Methods for evaluation of performance of personnel when dealing with severe accident management should be developed.

## POSSIBLE DRAWBACKS OF DEPRESSURISATION

Depressurisation of a reactor coolant system by bleed means extraction of energy at a higher rate than necessary for decay heat removal. If the system is not fed by low temperature coolant at the same rate as that at which coolant is being bled, the system inventory will decrease. During the blow-down, the flow of coolant through the core keeps the temperature down. After depressurisation, or when blow-down is terminated, the inventory could be insufficient to cool the core, leading to the start of fuel temperature increase. In this sense, depressurisation is not an optimal use of a limited coolant inventory. In a situation when the core already is overheated and clad oxidation has started, depressurisation could be insufficient to bring the pressure down sufficiently to meet the objectives of the action. Instead it could increase the generation of hydrogen and also the release of radioactive material into the containment. If the accident develops and core material slumps down into the water in the vessel lower plenum, any fuel-coolant interaction, if it occurred, may be more violent at lower system pressure.

However these possible drawbacks have to be assessed against the risks associated with vessel melt-through at high pressure.

## E. CONCLUSIONS

### GENERAL

There are circumstances, when failures in engineered safety systems give rise to serious difficulties in maintaining core cooling, in which intentional depressurisation of the primary circuit can confer significant benefits. First, depressurisation can delay, limit or even avoid altogether core degradation by facilitating recourse to alternative means of cooling and extending the time available for restoring the safety function which has failed.

Secondly, even if core degradation cannot be avoided, depressurisation can be used to mitigate the potentially severe consequences. It is clear that the severity of the consequences is directly and strongly dependent on the pressure within the RPV when it ruptures. Reducing the pressure prior to failure will lessen the risk of losing containment integrity and preserve the capability to continue implementing accident management measures.

In principle it is possible to achieve intentional depressurisation in all LWR reactor types using means incorporated in the original design, though there are differences specific to the characteristics of individual designs, particularly for PWRs. Blow-down valves on primary and secondary circuits are of dominant significance in that respect.

Intentional depressurisation is an operation which calls for well thought out operating procedures, defined in advance in relation to specific types of situation and validated. Successful implementation will depend on operator training based on "symptoms" and "signposts" clearly established to enable the operator to act in time without having to hesitate. Indeed, human factors are an essential element in the success of the operation. Whatever accident progression develops, at each possible stage procedures must be established for both the depressurisation itself and for subsequent actions to restore the plant to a safe state. It is believed that this can be done with adequate assurance.

An overriding requirement is that even if procedures do not succeed in their objective, at least they must not result in a significantly worse situation than would have occurred if they had not been implemented. So far no-one has identified generic examples where that would happen.

## SPECIFIC POINTS

On the whole, current understanding of phenomena directly related to intentional depressurisation seems adequate. However, attention is drawn to a few specific areas, such as flow regimes in valves and the detailed processes involved in injecting water with a view to restoring the situation, where additional R&D effort needs to be applied.

Finally the importance is stressed of qualifying valves and other relevant equipment to the degree appropriate to the specific conditions encountered in the course of intentional depressurisation.

February 1990

## ANNEX

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