IRSN
INSTITUT
DE RADIOPROTECTION
ET DE SÛRETÉ NUCLÉAIRE

Enhancing nuclear safety

Consideration of BORAX-type reactivity accidents applied to research reactors

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Consideration of BORAX-type reactivity accidents applied to research reactors

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IRSN

//in brief

The French Institute for Radiological Protection and Nuclear Safety (IRSN) was founded by **Act No.2001-398** of May 9, 2001 enacted through **Order No. 2002-254** of February 22, 2002. This Order was amended on April 7, 2007 to take into account the **Act No.2006-686** of June 13, 2006 relative on transparency and nuclear safety. The IRSN is a public establishment that carries out both industrial and commercial activities. It is jointly supervised by the Ministers for Defence, Environment, Industry, Research and Health.

IRSN employs over than 1,700 specialists: engineers, researchers, doctors, agronomists, veterinarians, technicians, experts in nuclear and radiation risks.

The Institute performs expert assessments and conducts research in the following fields:

- nuclear safety;
- safety relative to the transportation of radioactive and fissile materials;
- protection of human health and the environment from ionizing radiation;
- protection and control of nuclear materials;
- protection of facilities and transports dealing with radioactive and fissile materials against malicious acts.

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Foreword

The Institute for Radiological Protection and Nuclear Safety develops research programs and conducts studies on nuclear and radiological risks. It is responsible for public service initiatives aimed at prevention and provides technical support to the public authorities in charge of ensuring nuclear safety and security, together with radiological protection. In fulfilling these various duties, the Institute is called upon to define its position on certain scientific and technical issues.

In line with its policy of transparency and its desire to make quality information available to all partners and stakeholders for use in developing their own views, the IRSN publishes "reference documents", which present the Institute's position on specific subjects.

These documents are drafted by IRSN specialists, with the help of outside experts if necessary. They then undergo a quality assurance validation process.

These texts reflect the Institute's position at the time of publication on its website. It may revise its position in light of scientific progress, regulatory changes or the need for more in-depth discussion to satisfy internal requirements or external requests.

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We welcome your comments. These may be sent to the address given in the margin above and should include the reference to the relevant document.

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1/ Introduction

Most of the research reactors discussed in this document are pool-type reactors in which the reactor vessel and some of the reactor coolant systems are located in a pool of water. These reactors generally use fuel in plate assemblies formed by a compact layer of uranium (or U_3Si_2) and aluminium particles, sandwiched between two thin layers of aluminium serving as cladding. The fuel melting process begins at 660°C when the aluminium melts, while the uranium (or U_3Si_2) particles may remain solid.

The accident that occurred in the American SL-1 reactor in 1961, together with tests carried out in the United States as of 1954 in the BORAX-1 reactor and then, in 1962, in the SPERT-1 reactor, showed that a sudden substantial addition of reactivity in this type of reactor could lead to explosive mechanisms caused by degradation, or even fast meltdown, of part of the reactor core. This is what is known as a "BORAX-type" accident.

The aim of this document is first to briefly recall the circumstances of the SL-1 reactor accident, the lessons learned, how this operational feedback has been factored into the design of various research reactors around the world and, second, to describe the approach taken by France with regard to this type of accident and how, led by IRSN, this approach has evolved in the last decade.

2/ Accident at the SL-1 reactor in 1961

//Key lessons learned

2/1 Recap of the accident at the SL-1 reactor

The SL-1 reactor ("Stationary Low Power Reactor Number One") was a US Army experimental reactor built at the Idaho National Laboratory[1] site (40 miles west of Idaho Falls) as part of a programme to develop nuclear power plants for supplying electricity to remote sites, such as surveillance radar stations. It was commissioned on 11 August 1958. The reactor's maximum thermal power was 3 MW. It was capable of electricity output of 200 kW. The reactor core consisted of approximately a hundred aluminium- and uranium-based plates, clad in aluminium and grouped in fuel assemblies; the fuel was manufactured at the Argonne National Laboratory. The uranium was enriched to 93 % in isotope U-235. The reactor had nine cadmium-based absorber rods (or control rods). The water in the (closed) reactor vessel served both as coolant and neutron moderator. Figure 1 at the end of this document shows a cross-section view of the SL-1 reactor.

[1] INL: "Idaho National Laboratory"

At the end of December 1960, it was decided to carry out maintenance on the absorber rods, following several occasions on which they had become jammed. The reactor was shut down to carry out this maintenance; the rods were placed in the low position and disconnected from their control mechanisms. On the afternoon of 3 January 1961, once the maintenance operations were complete, a team reconnected the drive mechanisms to their respective rods in order to restart the reactor.

At 9 pm, alarm signals from the SL-1 reactor building sounded at three fire stations. These alarms did not make it clear whether the problem was a fire or an abnormal radiation level. Upon their arrival at the site, the emergency response teams could not detect any visible signs of damage or fire. However, very high irradiation dose rates were detected at the entrance of the reactor building, with values of around 1,000 rad/hour (10 Gy/h) in the reactor hall. Two people were found motionless near the reactor, while a third person had been projected and pinned to the roof of the building by one of the absorber rods. Two of the three men were killed instantaneously, while the third would die two hours after the accident, en route to hospital.

The inspections performed, notably those using a robot, led to the conclusion that only the central absorber rod had been ejected; the other rods had remained inside the core, which had suffered major radial deformation. A radiological protection plug had been ejected up to the ceiling of the building. The state of the core can be seen in Figure 2 at the end of this document. The vessel and the reactor building both withstood the accident.

The most widely-accepted theory regarding the cause of the accident is that an absorber rod got stuck and one of the operators decided to free it manually, but withdrew it too far. The rod was raised too high, exceeding the limit above which the chain reaction becomes uncontrollable, thus leading to the reactor explosion. Mainly due to the presence of a short-lived yttrium isotope found on the dead operators' clothing, it was estimated that reactor power may have surged to around 20,000 MW[2] during the accident; given the damage observed, vessel pressure may have exceeded 30 bars.

[2] And a total of 1.5 x10^{18} fission reactions.

It took over a year to decontaminate the SL-1 reactor; all the debris from the reactor was removed and the building was demolished in 1962.

The members of the rescue crew who were the most exposed received an estimated dose of 30 rad (0.3 Gy). There were no significant radiological effects outside the building, where 99.99 % of the radioactivity was contained. Downwind of the SL-1 reactor, the radiological impact on plants remained low and no groundwater contamination was detected.

The data available regarding this accident shows that, in addition to ensuring radiological protection for the crews, one of the main concerns of the people in charge of organising the emergency response was to avoid any risk of a second nuclear accident, by checking that there were enough absorber rods in the reactor core and that there was no danger of the ejected plug falling back down on top of the reactor.

A number of publications relative to the SL-1 reactor accident and BORAX-type accidents in general are available on the Idaho National Laboratory's website, at www.inl.gov/proving-the-principle.

2/2
Key lessons learned from the SL-1 reactor accident

The accident that occurred in the SL-1 reactor, together with tests carried out in the United States as of 1954 in the BORAX-1 reactor[3] then, in 1962, in the SPERT-1 reactor[4], which are discussed in more detail here after, showed that a sudden substantial addition of reactivity in water-cooled research reactors that use aluminium- and uranium-based metallic fuel could lead to the degradation or even fast meltdown of part of the core (with the possibility of these two phenomena occurring simultaneously):

- water vaporisation (steam explosion), by thermodynamic interaction between the molten materials and the coolant water; aluminium oxidation can lead to a significant excess of thermal energy in the coolant;
- possibly, the sudden vaporisation of the aluminium.

These phenomena can generally lead to shock waves and to the collapse of bubbles in the reactor coolant and – in the case of pool-type reactors - in the pool. These bubbles may contain noncondensable gases (hydrogen resulting from the oxidation of aluminium or damaged experimental devices, etc.) which are liable to amplify the mechanical effects of the dissipation of steam bubbles.

Considered in greater detail, this type of accident may:

- damage the reactor coolant system and the pool walls;
- damage the lower part of the containment (bottom of the pool), due to the thermal effects of non-ejected molten materials;
- induce a water transfer in the hall of the reactor building, following the steam explosion, liable to impact the containment;

[3] "BOiling water ReActor eXperiment" — See Figures 3 and 4 at the end of this document, showing a cross-section of the BORAX-1 reactor and a photograph taken during the "destructive" test.

[4] SPERT-1: "Special Power Excursion Reactor Test"

- cause the temperature and air pressure in the reactor building wall to rise in the long term, due in particular to the transfer of noble gases and volatile fission products within the containment, possibly carrying particles or fragments of fuel;
- very high dose rates inside the reactor building;
- radioactive releases into the environment.

On the other hand, assessing the consequences of a BORAX-type accident, such as the transfer of radioactive products inside the reactor building, radioactive release into the environment, post-accident cooling of the melt and the risk of "recriticality", are not dealt with in this document, even though they are involved in the safety demonstration.

3/

Consideration of fuel melting accidents in research reactors

//How France’s approach has changed

3/1
Consideration of fuel melting accidents in research reactors

Table 1[1] at the end of this document shows a series of water-cooled research reactors using aluminium- and uranium-based fuel, in chronological order of commissioning. Fuel melting was considered in design studies for most of these reactors, including the earliest listed in the table (HIFAR, commissioned in 1958). It should be noted that:

- total core meltdown was not systematically considered (total meltdown in the case of certain reactors, melting of a few fuel plates in other cases);
- melting is not always the result of a reactivity accident; in some cases, it is related to loss of reactor core coolant, which affects the sequence and consequences of an accident due to the progressivity of the melting;
- for cases where meltdown is due to a reactivity accident, the thermal energy[2] considered in the core may be as high as 200 MJ (as in the case of the BR2 reactor at the Mol Centre in Belgium);

[1] This table is based on information gathered by IRSN.

[2] Thermal energy means all the nuclear power released in the fuel during the reactivity transient.

- from the data gathered, it would seem that reactivity accidents with no explosive interaction between the fuel and water were only considered in the case of French reactors and Belgium's BR2 reactor. The possibility of a steam explosion occurring was not considered for the core meltdown accident at the FRM-II reactor in Germany. The justification given for this is that it is impossible for a reactivity insertion in this reactor to lead to a "neutron rate"[3] low enough to induce an explosive reaction with the water;
- meltdown accidents "in air" were considered for French research reactors (melting of an assembly during a handling operation). This case entails slow melting due to loss of coolant (through residual heat).

3/2

How France's approach has changed

In France, the possibility of a BORAX-type accident occurring has always been taken into consideration in containment design[4] for water-cooled research reactors operated using aluminium- and uranium-based metallic fuel.

To take this type of accident into consideration, measures are defined to guarantee a high prevention level with regard to events that may initiate reactivity insertion[5] in the core, and the consequences of such an accident occurring at the facility are assessed to check compliance with the functional requirements specified in such a situation for the various facilities that make up the containment system (reactor building, liners and pools, ventilation and filtration systems, and post-accident cooling systems, etc.). Robust containment design is required with regard to the BORAX-type accident because there are generally no design measures available to reduce the impact of core uncovery and loss of containment during this type of accident.

The initiating events considered are the ejection of one or more control rods, reactivity insertion in the event of an experimental absorber device being removed, etc.

The consequences of a BORAX-type accident have been assessed in the design of French research reactors according to an "inclusive" approach, without studying different scenarios relative to reactivity addition in the core. The key characteristics of this approach were 135 MJ of thermal energy deposited in the fuel and mechanical energy, from the thermodynamic interaction between the molten materials and the coolant, representing 9 % of thermal energy;

[3] See definitions for Note 2, Table 2 at the end of this document.

[4] The term "containment" refers here to the third containment barrier of the reactors addressed in this document. It consists of the containment building (i.e. the "reactor hall" building) at the top and the pool at the bottom, together with the related systems and equipment (ventilation, discharge filtration systems, etc.). Design covers all the studies aimed at providing a precise definition of equipment functional requirements and technical characteristics (thickness of a concrete wall, reinforcement ratio, pump flowrate, vessel thickness, materials used, selected weld types, etc.).

[5] The term "insertion" and "addition" are generally used in reference to reactivity contributions in a reactor.

these characteristics are those adopted for the most recent research reactors built in France, namely, the High Flux Reactor (HFR*)* in Grenoble and the ORPHEE reactor in Saclay. For these reactors, the 135 MJ energy represented 100 % of the molten core, heated to a temperature of about 800°C. At the time of design, it was considered that the values mentioned above were "envelope" values, based on analysis of the SL-1 reactor accident and tests carried out on the BORAX-1 and SPERT-1 reactors (see Table 2 at the end of this document).

Additionally, at the stage of drawing up the safety options file for the Jules Horowitz reactor (which contains more fuel than OSIRIS, the HFR and ORPHEE - see Table 1), the designer proposed applying the following values: 135 MJ of thermal energy in the fuel and 6.75 MJ of mechanical energy, i.e. 5 % of the thermal energy in the fuel. In the case of the Jules Horowitz reactor, the 135 MJ energy was equivalent to a 50 % core meltdown.

These proposals have been claimed by IRSN, on the basis of elements hereafter.

4/

Policy considerations

It should first be stressed that BORAX-type accidents appear to be considered in the design and safety analysis of water-cooled research reactors operated using aluminium- and uranium-based fuel, given that the occurrence of reactivity accidents is physically possible:

- in view of the purposes for which they are built, the reactors in question are facilities that can be used – often simultaneously - for a variety of ends, such as carrying out experimental programmes and producing radioisotopes, etc.; building and loading assemblies, and even whole cores, "to order" entails a great deal of handling;
- the life of these reactors is often complex, sometimes involving changes to their purposes and facilities over the course of time; the experiments carried out require dedicated support systems liable to generate certain risks that are not always explicitly planned for at the time the reactor is designed (pressurised gas, etc.), even if they are subject to safety analysis and authorisation procedures; consideration of a BORAX-type "envelope" accident at the design stage makes it easier to implement such changes at a later date;
- these reactors use specific equipment for which there is no reliability data or little available operating feedback;
- for such reactors, human and organisational factors play a very specific role in incident and accident prevention; even if lessons have been learned from the SL-1 accident and from major accidents at power reactors (Three Mile Island, Chernobyl and soon Fukushima), the possibility of human error and the chances of recovering from such errors are not

easy to quantify; neither is it easy to evaluate how robust organisational “defence lines” are, which gives rise to a great deal of debate; lastly, the coexistence of two interacting categories of personnel (operators and experimenters) during experimental stages also creates a particular risk.

Furthermore, the technical basis proposed during the drafting of the safety options file for the Jules Horowitz reactor led IRSN to review various aspects of the BORAX-type accident in 2003 and to include it in safety studies:

- conditions leading to a steam explosion;
- determination of the thermal energy deposited in the fuel during the accident;
- assessment of the mechanical energy and pressure that can build up as a result of the thermodynamic and chemical interaction between the coolant and the molten part of the core;
- assessment of thermomechanical loads and damage to containment structures (shock waves on pool structures, water ejection effect in the containment building, etc.);
- assessment of radiological consequences.

It should be emphasised that assessing the various consequences of such an accident is complicated and that there is no overall computational tool available that can be used in particular for modelling all the reactor structures (reactor vessel or tank with closure structures, reactor coolant system, reflector, experimental systems, pool liner, concrete walls of the pool, etc.), as well as all the different phenomena involved. During the design stage for any new reactor, however, it is essential to reach a decision as to the adequacy of the fundamental hypotheses applied for the design of components involved in containing any radioactive products released during the accident (especially thermal and mechanical energy). One possible approach, which has been used for other reactors[1], consists in implementing special assessment tools for each of the targeted consequences of the accident, in order to estimate the undesirable effects with safety margins, while remaining realistic. For example, in the case of the BORAX-type accident, we need to assess the risk relative to a water ejection that could immediately cause overpressure in the containment due to heat transfer between the water and the air; in this case, a set of

[1] For the sodium –cooled fast reactor, SUPERPHENIX, for example.

realistic hypotheses can be applied to increase the quantity of water liable to be ejected inside the containment. Another set of hypotheses, related, as appropriate, to another computational tool, may be used to assess the risks related to the effects of the shock wave against the reactor vessel or pool walls, adopting a conservative approach to the effects. The same reasoning can be applied in assessing the radiological consequences for the population and the environment (direct irradiation via the containment, direct leaks through the containment, indirect and filtered leaks).

4/1 Triggering a steam explosion

The hypothesis of a steam explosion triggered by the thermodynamic interaction between molten materials and the coolant water[2] is considered in the three reports quoted in References 1, 2 and 3, relative to the destructive tests performed in the BORAX-1 and SPERT-1 reactors and to the SL-1 reactor accident.

[2] The terms steam explosion and thermodynamic interaction between the molten fuel and the coolant water are used without distinction.

Based on all the interpretations of the tests and accident mentioned above (and summarised in a communication dating back to 1989, quoted in Reference 4), the designer of the Jules Horowitz reactor initially considered that there was a threshold, corresponding to a reactivity insertion with a "neutron rate" of 4 ms, below which a steam explosion was possible; this threshold could then have been used directly to define reactivity insertions liable to lead to a BORAX-type explosive accident in a reactor.

The principle on which this approach is based is questionable, since it is not based on thermodynamic considerations. A steam explosion can occur as a result of contact between two liquids, one of which, the fuel, is extremely hot, while the other, the coolant, is cold and volatile. IRSN considers that it is appropriate to adopt a thermodynamic approach to determine the conditions that may trigger a steam explosion; this assumes that a study is made of the reactivity transients that may lead to partial or total core meltdown.

Moreover, the conclusion to the report on the tests performed in the SPERT-1 reactor, clearly states that, while the reactivity added, the reactor's "neutron rate" and feedback effects allow assessment of the thermal energy deposited in the fuel during the transient, they cannot be used to predict whether or not an explosive thermodynamic interaction will be triggered. The fraction of molten fuel and the maximum temperature reached in the fuel clearly appear in the report's conclusions as important factors in triggering a steam explosion.

IRSN therefore considers that it is not currently possible to define a precise explosion criterion, given that the triggering of a steam explosion is a complex phenomenon. Although experimental observations demonstrate that, generally-speaking, an explosive thermodynamic interaction between molten fuel and the coolant does not occur systematically, it does seem necessary to consider the possibility of an explosive accident during the design phase of water-cooled research reactors operated using metallic fuel, if molten materials can suddenly come into contact with coolant in the liquid state.

4/2 Thermal energy deposited in the fuel

The thermal energy deposited in the fuel during the accident is a standard parameter for determining the extent of the steam explosion.

The experimental data relative to the BORAX-type accident, which mainly comes from the SL-1 reactor accident and the BORAX-1 and SPERT-1 reactors (see Table 2), reveals the following key points:

- no phenomenon occurs that limits the amount of thermal energy deposited in the core during the accident: this energy is heavily dependent on the reactivity inserted, the kinetics by which it is inserted, neutron feedback and the quantity of fuel in the core, etc. The three reports on the BORAX-1, SPERT-1 tests and the SL-1 accident (quoted in references 1, 2 and 3 respectively) do not include any data that can be used to assert that 135 MJ is an envelope value. The report describing the tests in the BORAX-1 reactor is especially interesting in this respect: it shows how the key characteristics of the test that resulted in the destruction of the reactor were defined on the basis of correlations based on preliminary test results. The objective of the "destructive" test was to reach the point where the fuel elements begin to melt and it was determined, through extrapolation based on the preliminary test results, that such an objective could be reached by gradually injecting reactivity equal to a "neutron rate" of between 2 ms and 2.5 ms and thermal energy of 80 MJ. The difference between the amount of energy expected (80 MJ) and that actually deposited insofar as it could be evaluated, i.e. 135 MJ, may be due to uncertainty related to the extrapolations carried out. A target thermal energy higher than 80 MJ could have been defined and, under the same test conditions, energy higher than 135 MJ would then most likely have been deposited in the fuel;

- the analyses carried out tended to indicate that the main neutron feedback effects which limited the power peak were fuel expansion due to the rise in temperature (the Doppler effect is marginal in the case of highly-enriched fuel), together with the moderator density effect when the temperature increases (see the two communications quoted in References 5 and 6). It should be noted that the fuel involved was highly enriched in 235U (93 %), which is not the case for all research reactors, nor for the Jules Horowitz reactor in particular, which will use lower-enriched fuel where the Doppler feedback effect will limit foreseeable power excursions;
- the reports mentioned above reveal the influence of some characteristic parameters of the core on the accident sequence (and thus in the thermal energy deposited in the fuel): kinetic parameters, moderator and Doppler feedback coefficients, and the geometric characteristics of the fuel plates, etc. Care should therefore be taken when extrapolating the results of the BORAX-1 or SPERT-1 tests in the case of a power reactor;
- the additional insertion of thermal energy due to chemical reactions seems to be of secondary importance in the aforementioned tests and during the SL-1 reactor accident; nonetheless, oxidation of the aluminium by water speeds up above 1,170°C, and even more so above 1,750°C, and although these values were not reached during the BORAX-1 and SPERT-1 tests, they could be reached in other configurations (regarding the reactivity insertion or the core's neutron characteristics, etc.). The potential addition of energy due to aluminium oxidation is considerable (15 MJ per kilogram of aluminium).

To conclude, the review did not find any data to support adopting a value of 135 MJ as the limit for the thermal energy deposited in the fuel during a BORAX-type accident.

In view of the above, IRSN now considers that the approach adopted should include the identification and study of different scenarios representative of possible reactivity addition sequences, drawing on the state of the art (knowledge and models available relative to the various phenomena involved) and making use of developments in numerical simulation to ensure that the thermal energy value ultimately adopted really does encompass all the values likely to be encountered during these sequences; to this end, the following aspects must be studied:

- reactivity that can be inserted in the reactor core;
- thermal energy deposited in the core at the end of the reactivity insertion transient.

It should be noted that the use of simulation tools in assessing fast reactivity insertion transients in research reactors implies the availability of sufficient quantities of data relevant to the cases studied. In this respect, and based on currently available knowledge, some gaps exist, such as those concerning thermal-hydraulic correlations for processing fast transients.

4/3 Thermal energy transferred to water

When designing the reactor containment, it is necessary to calculate the mechanical energy that a steam explosion may generate. This mechanical energy depends primarily on the thermal energy transferred from the molten fuel to the water. The resulting build-up in pressure (explosion) is caused by the water boiling. The more powerful the explosion, the faster the energy transfer. The speed of thermal energy transfer is essentially due to the fuel breaking up into fine fragments (pulverisation), this phenomenon in turn being induced by the sudden rise in pressure (“pressure shock”). Although the mechanisms are clear, the quantity of energy transferred remains uncertain. This can be calculated using dedicated computer codes, such as MC3D in which law and parameter have been defined as precise as possible considering the state of the art and the experimental backup in particular. The main uncertainty affects the power of energy transfer and fragment size.

In view of the shortage of experimental support available on the subject especially for aluminium-based fuel, sensitivity studies should be carried out for determining an "envelope" value of the energy transferred to water.

4/4
Induced mechanical effects

A steam explosion caused by thermodynamic interaction between molten materials and water can produce mechanical work, as in the case of the destructive tests carried out in the BORAX-1 and SPERT-1 reactors, and during the SL-1 reactor accident. The related reports quoted in References 1, 2 and 3 give a mechanical efficiency value corresponding to the ratio between mechanical energy, which is largely estimated from the deformation and damage observed in structures and equipment, and the thermal energy deposited in the fuel.

The mechanical energy induced by a steam explosion, bounded by the value of the thermal energy transferred from the molten fuel to the water, can be seen in two mechanisms: shock wave propagation and the setting into motion of water by the thrust from steam bubbles[3]. These phenomena can cause:

- the deformation - or even rupture or breakage – of structures and equipment, including metal enclosures around the core, upper core plate and reactor coolant system piping, experimental devices installed at the edge of the core, equipment located in the pool, pool liner, gate between the pool and transfer canal, etc.
- the ejection of water in the reactor building hall.

In this respect, the use of mechanical efficiency values directly based on the analysis of the tests performed in the BORAX-1 and SPERT-1 reactors and of the SL-1 reactor accident (which is already subject to significant uncertainties) seems highly questionable, since these reactors are not particularly representative of pool-type reactors. Figure 1 shows that in the case of the SL-1 reactor, the core environment was relatively "stressed" in comparison with the case of a pool-type reactor, as illustrated in Figure 5. It should be emphasised that, ultimately, no steam bubbles would be created and no mechanical energy released if molten material were to come into contact with an incompressible liquid in an absolutely rigid

[3] The work associated with the collapse of steam bubbles represents part of the mechanical energy associated with the steam explosion.

environment. The energy transferred to the liquid would only heat it. For this reason, mechanical efficiency is a difficult concept to deal with.

For each reactor, particularly close attention must therefore be paid to the possible mechanical effects of foreseeable thermodynamic interactions between molten material and water, considering factors such as the hydrostatic pressure of pool water, volume and inertia of the bodies of water likely to be affected, stiffness and inertia of internal metal structures and pool lining, etc. In addition, the rigidity of the "environment" of the interaction area is not necessarily uniform in every direction. As a result, water may be set into motion in a preferential direction - such as upwards ("cannon effect").

A computer code, such as MC3D, can be combined with carefully adapted models to simulate the above phenomena.

Given the complexity of these phenomena and the related uncertainties, however, IRSN considers that a complementary experimental approach, based on suitable mock-up tests, remains necessary. It should be pointed out that explosion tests have been carried out on small-scale mock-ups of reactors, including OSIRIS, the HFR and ORPHEE in particular. The main purpose of these tests was to confirm the calculations relative to pool behaviour (concrete and metal liner). The tests used specially adapted explosive charges to simulate a steam explosion.

5/ Conclusion

The accident at the SL-1 research reactor on 3 January 1961, together with tests carried out in 1954 at the BORAX-1 reactor and in 1962 at the SPERT-1 reactor, have demonstrated that, in the event of a sudden and substantial addition of reactivity in the core of reactors that use aluminium- and uranium-based fuel, a violent release of energy can occur, which may cause the partial or total destruction of the reactor core and nearby structures.

In France, this type of reactivity accident, known as a BORAX-type accident, is systematically considered in designing containment systems (reactor containment, ventilation and filtration systems, etc.) for reactors that use aluminium- and uranium-based fuel, all of which are water-cooled. IRSN considers that this is still necessary, even if lessons have been learned from the accident at the SL-1 reactor, and, more generally, from other accidents that have occurred at nuclear reactors, in order to enhance prevention of a massive and sudden insertion of reactivity into research reactors.

IRSN also recommends reconsidering the approach previously adopted for this type of research reactor, built in France as from 1966, which takes into account an unique value of the thermal energy deposited in the fuel (135 MJ). Instead, the thermal energy value considered should take into account:

- the possibilities of reactivity insertion in the core;
- specific reactor properties, including the amount of fuel in the core, the enrichment of the fuel in uranium-235 and neutron feedback coefficients;
- aluminium oxidation reactions.

Obtaining this thermal energy value, however, is not sufficient for designing reactor containment systems. It is also necessary to determine the mechanical energy likely to be produced as a result of heat transfers from the molten fuel to the water.

In this respect, IRSN considers that the thermodynamic interactions between molten fuel and reactor coolant should be examined for all scenarios that include the possible melting of the fuel in the presence of water in the liquid state. IRSN also considers that: a) care should be taken to determine the mechanical energy associated with the thermodynamic interaction, allowing for the specific characteristics of the reactor in question (bodies of water involved, stiffness and inertia of nearby structures); and b) the use of "mechanical efficiency" values found in the "literature" may be inappropriate.

The approach set out above must lead to a clear understanding of the conditions to which the reactor containment may be exposed. It must also allow the definition of the thermomechanical loads to be considered for designing and sizing reactor containment structures, making allowance for calculation uncertainties and the potential impact of phenomena which are not modelled (whether these concern the evaluation of thermal energy deposited in the fuel, energy transfers to the water or induced mechanical effects). Nonetheless, given the state of the art in simulation, the complexity of the phenomena to be modelled and the related uncertainties, suitable mock-up tests would seem necessary to confirm assessments based on calculations.

6/

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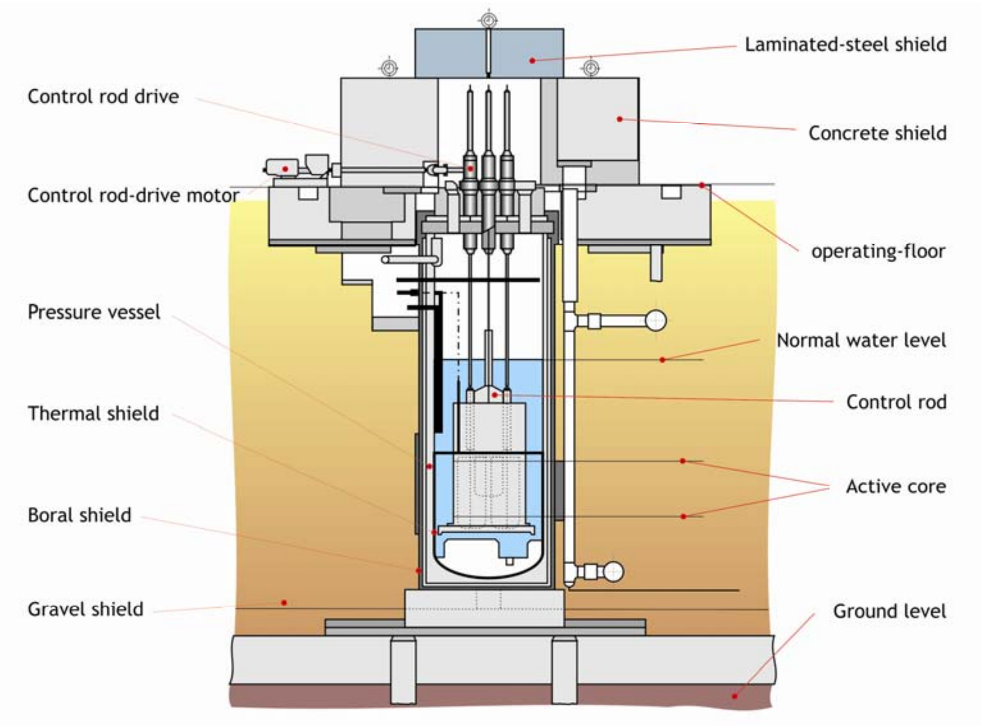

Figure 1.
Cross SL-1 reactor.

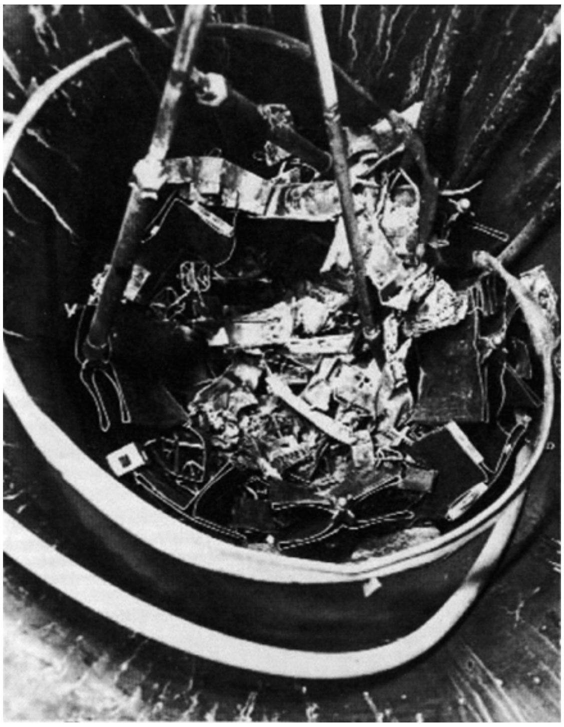

Figure 2.
View of the SL-1 reactor core following the 1961 reactivity accident - Three control rod drive mechanisms are visible.
Photo credit: INL

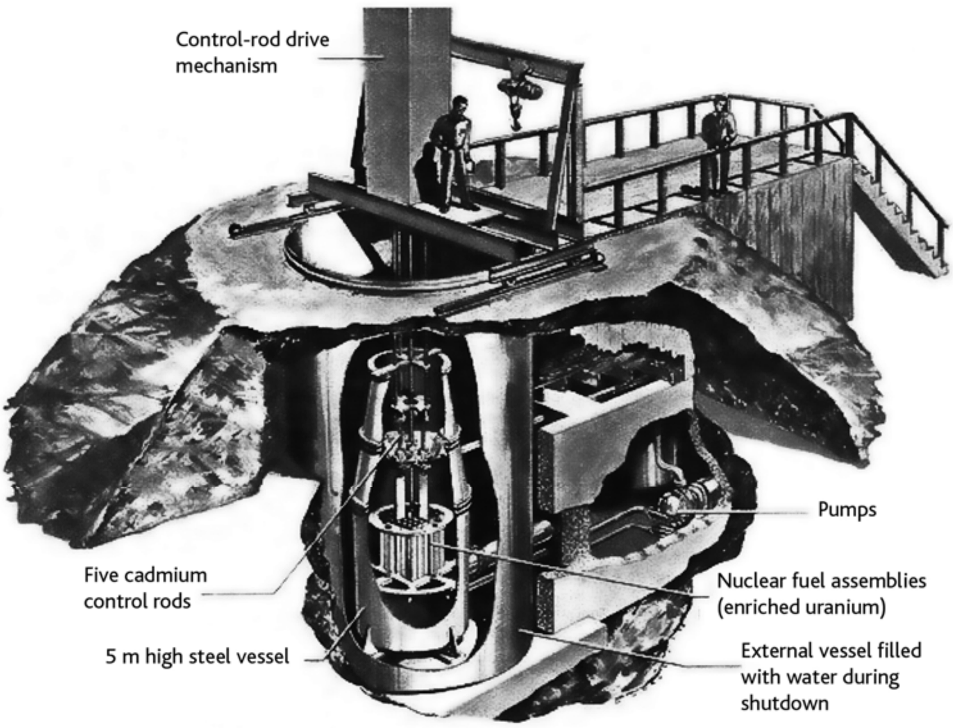

Figure 3.
Cross-section of the BORAX-1 reactor.
Photo credit: DR

Figure 4.
Photography taken during the "destructive" test on the BORAX-1 reactor.
Photo credit: INL

Figure 5.
Simplified diagram of a pool-type reactor.

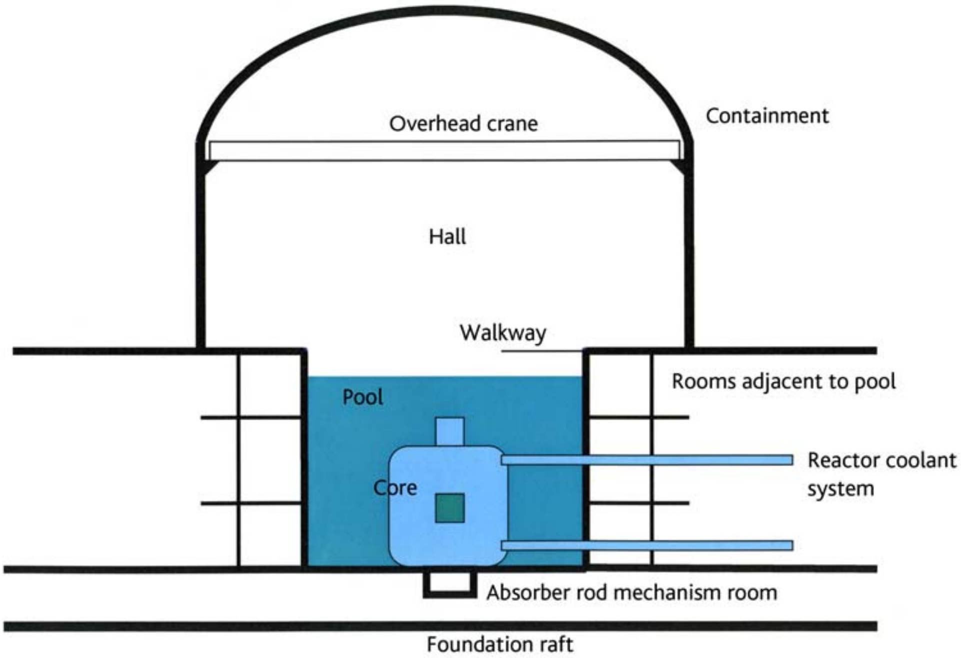

Table 1.
Fuel melting accidents considered for different experimental reactors.

Reactor Year commissioned – Year decommissioned	Country	Power (MW)	Fuel	Fuel melting accidents considered *(fusion under water, unless otherwise stated)*
HIFAR 1958-2007	Australia (Lucas Heights)	10	U-Al enriched to approx. 60 % in ^{235}U	Total core meltdown caused by loss of coolant Total core meltdown and containment failure
HFR 1961	Netherlands (Petten)	50	Initially U-Al enriched to 91 % in ^{235}U, then U_3Si_2-Al enriched to approx. 20 % in ^{235}U	No melting accidents Reactivity accident is impossible because the direction of water flow prevents absorber ejection Blocked water channel in a fuel assembly: does not lead to melting
BR2 1963	Belgium (Mol)	100	U-Al enriched to approx. 93 % in ^{235}U	Power excursion leading to core meltdown, followed by interaction between the aluminium and water — 200 MJ
SAFARI-1 1965	South Africa (Pelindaba)	20	U-Al enriched to 87 %-93 % in ^{235}U	Total core meltdown accompanied by failure of ventilation systems
OSIRIS 1966	France (Saclay)	70	U_3Si_2-Al enriched to approx. 20 % in ^{235}U	Power excursion leading to total core meltdown, followed by interaction between the aluminium and water — 135 MJ Melt of a fuel assembly “in air”, during a handling operation
HFR 1971	France (Grenoble)	57	U-Al enriched to 93 % in ^{235}U (total aluminium: 60 kg)	Power excursion leading to total melting of the core (which contained only one fuel assembly), followed by interaction between the aluminium and water — 135 MJ Total uncovery and melting of the core “in air”
ORPHEE 1980	France (Saclay)	60	U-Al enriched to 93 % in ^{235}U (total aluminium: 57.5 kg)	Power excursion leading to total core meltdown, followed by interaction between the aluminium and water (3) — 135 MJ Melt of a fuel assembly “in air”, during a handling operation (3)
RSG-GAS 1987	Indonesia (Serpong)	30	U_3Si_2-Al enriched to approx. 20 % in ^{235}U	Melt of a fuel assembly due to a blocked channel (1) Transient with postulated failure of reactor protection system (ATWS[1]) leading to melting of 5 fuel assemblies (2)
FRM-II 2004	Germany (Garching)	20	U_3Si_2-Al enriched to approx. 90 % in ^{235}U	Total core meltdown caused by loss of coolant or a reactivity accident, with failure of reactor protection system (2) – No interaction between aluminium and water
OPAL 2007	Australia (Lucas Heights)	20	U_3Si_2-Al enriched to approx. 20 % in ^{235}U	36 UMo “targets” melted, caused by loss of coolant (2) 3 fuel plates melted due to a partially blocked fuel assembly channel (2)

[1] ATWS: « Anticipated Transient Without Scram »

Table 1 (next)

Reactor **Year commissioned – Year decommissioned**	**Country**	**Power (MW)**	**Fuel**	**Fuel melting accidents considered** ***(fusion under water, unless otherwise stated)***
JHR Under construction	France	100	U_3Si_2-Al enriched to approx. 27 % in ^{235}U (aluminium: 114 kg) (later replaced by UMo-Al enriched to approx. 20 % in ^{235}U)	Blocked water channel in a fuel assembly (1) At the stage of drawing up the safety options file, 50 % of the core melted under water with aluminium-water interaction, thermal energy deposed - 135 MJ *Note: uncovery of a fuel assembly, total core meltdown "in air": explicitly excluded events (appropriate prevention)*
CARR First divergence in May 2010	China	60	U_3Si_2-Al enriched to approx. 20 % in ^{235}U	Clogging in some channels in a fuel assembly (1) Melt of three fuel assemblies (2) considered for drawing up the On-site Emergency Plan

In light of data currently available to IRSN, the status of the different accidents listed in this table is not always very clear. This status is only mentioned where there is no ambiguity involved, mainly in the case of the most recently-built reactors:

(1) accident considered in the "design-basis" domain according to the practice adopted in France for reactor design and safety demonstration. This notion of domain entails the study of various operating conditions corresponding to stable and normal transient operating states, incidents and accidents. They are divided into four categories;

(2) accident considered within a "beyond design basis" framework (yet which serves in the design (including design basis) of certain equipment and systems used to limit the consequences, such as the reactor containment, ventilation and filtration systems and post-accident cooling system);

(3) as part of the recently completed safety review for the ORPHEE reactor, the operator has implemented a design and safety demonstration approach based on the study of operating conditions, as mentioned in note (1) above; the two accidents mentioned above are studied within a "beyond design basis" framework: see note (2).

Table 2.
Summary of BORAX-type power excursion tests (U-Al plate fuel with uranium highly enriched in Uranium-235).

Reactor	Description	Total mass of Al and U (kg)	Reactivity insertion[1] ($)	T[2] (ms)	Power spike (MW)	Thermal energy released (MJ)	Chemical energy released (MJ)	Maximum fuel temperature (°C)	Pressure spike (bar)	Damage to reactor core
BORAX-1	Destructive test	100/4.2	3.1	2.6	< 19,000	135	Undetermined	< 1,800	400 - 700	Extensive reactor core melting
SL-1	Accident	189/14	3.0	3.6	~ 19,000	133 ± 10	24 ± 10	> 2,075	700	approx. 20 % of core melted approx. 2 % of core vaporised
SPERT-1	Final non-destructive tests	51/3.8	2.6	5.0	1,130	11	-	585	0.5	> 0.5 % of core meltec
			2.7	4.6	1,270	19	-	680	0.5	approx. 2 % of core melted
	Destructive test		3.55	3.2	2,250	31	3.5	1,360	< 300	approx. 35 % of core melted

[1] The dollar sign ($) represents the proportion of delayed neutrons of the reactor as well as the quantity of reactivity which, if added when the reactor is in operation, leads to an uncontrolled chain reaction through the prompt neutrons alone.

[2] Reactor period (T): for a level of reactivity $\Delta\rho > \beta$, the neutron population increases exponentially
$n(t) = n_0 \exp(t/T)$ where
$1/T = (\Delta\rho - \beta) / l$
(asymptote in Nordheim equations), where:
l: half-life of prompt neutrons
β: fraction of delayed neutrons