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SITING OF RESEARCH REACTORS

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FOREWORD

Safety criteria for the siting of research reactors of a thermal power level up to a few megawatts are developed in this document. However, the general concepts can be extrapolated for use with research reactors up to a few tens of megawatts.

In order to grade safety requirements to different types of research reactors, a classification in three groups is proposed and general safety criteria are established for each group.

The main purpose of the document is to provide some guidance to Member States on the following three main topics: evaluation of the radiological impact of the reactor installation under normal and accident conditions, effects on the safe operation of the reactor of extreme natural events and man-induced external events, and emergency planning feasibility for each group of reactors.

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INTRODUCTION

The purpose of this document is to develop criteria for siting and the site-related design basis for research reactors. The concepts presented in this document are intended as recommendations for new reactors and are not suggested for backfitting purposes for facilities already in existence (see Annex 1).

In siting research reactors, as for all other nuclear installations, serious consideration is given to minimizing the effects of the site on the reactor and the reactor on the site, i.e. the effects of extreme external events, both natural and man-made (earthquake, floods, air crashes, chemical explosions etc.), and the potential impact of the reactor on the environment.

In this document guidance is first provided on the evaluation of the radiological impact of the installation under normal reactor operation and accident conditions.

A classification of research reactors in groups is then proposed, together with a different approach for each group, to take into account the relevant safety problems associated with facilities of different characteristics.

Guidance is also provided for both extreme natural events and for man-induced external events which could affect the safe operation of the reactor. Extreme natural events include earthquakes, flooding for river or coastal sites and extreme meteorological phenomena. Among man-made events consideration is given to events such as releases from industries or transport facilities which may lead to explosions, airplane crashes, releases of toxic gases etc.

The feasibility of emergency planning is finally considered for each group of reactors.

Because of the relatively small power, core size and fission product inventory of research reactors in comparison with power reactors, postulated accidents will normally result in relatively small releases of radioactivity to the environment. Only exceptionally large research reactors could possibly release an amount of radioactive material not too different than power reactors. It is not the purpose of this document to consider this type of reactor. The safety analysis for such reactors will always be a special case and should probably follow guidance documents developed for the siting and design of power reactors.

The specific criteria contained in this report therefore can be applied to research reactors as defined in IAEA Safety Series No. 35 "Safe Operation of Research Reactors and Critical Assemblies" [1], with the following conditions:

- The thermal power level is lower than a few MW (~5MW); however, the general concepts can be extrapolated to be used for research reactors up to a few tens of MW*;

- Fast reactors and cores with significant plutonium inventory are not considered.

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* It should be remembered that IAEA TEC-DOC 348 "Earthquake Resistant Design of Nuclear Facilities with Limited Radioactive Inventory" [2] which represents a basis for the present document in which the seismic effects are considered, can be applied only up to a few MW (~5MW). For power higher than this the methods of analysis to be used together with the results obtained should be approved by the Regulatory Body.

1. RADIOLOGICAL ASPECTS

1.1. Normal Operation

Although during normal operation of research reactors exposure of the public due to the release of radioactive materials in the environment is expected to be negligible, a site-related assessment of such exposure should be performed prior to operation. The main objective of this assessment is to demonstrate compliance with the system of dose limitation as described in detail in IAEA Safety Series No. 9 "Basic Safety Standards for Radiation Protection" [3]. These evaluations are normally required by regulatory authorities and include estimating the effective dose equivalent for the most exposed members of the public (critical group) and collective effective dose equivalent commitment of the population ("collective dose").

Environmental models can be used to establish relationships between discharges, environmental contamination levels and the resulting doses to the public. If the resulting doses are not negligible, site specific transfer factors should be determined on the basis of local studies.

Several parameters should be identified and quantified during the site survey and the pre-operational stages. These are the critical groups and the pathways leading to them and estimates of the relevant transfer factors along the pathways.

The quantity and nature of routine radionuclide releases may be estimated from the safety analysis of the proposed installation or from operating experience with reactors of design similar to the type planned for the site.

By using this information as an estimated "source term" and considering pathway factors an assessment of representative doses to the critical groups can be obtained.

In the late stages of site qualification it may only be necessary to make rough and conservative evaluations showing that expected doses in the potential critical group are well under the upper bounds stipulated for this type of installation by the competent authority.

1.2. Accident Conditions

The primary design objective of a nuclear facility against the related extreme event is to ensure an adequately low probability of damage to structures and equipment that could lead to significant exposure of plant personnel or members of the public. In the case of research reactors, the need for design against extreme events depends on the probability of the extreme event and on the consequences that such an event would have if no design features resistant to extreme events were applied.

A site-related design resistant to extreme events having a low probability of occurrence include the following objectives:

(i) To prevent radiological consequences from events equal to or smaller than the design basis;

(ii) to ensure that the risk incurred by the potentially exposed population from larger events is kept at a sufficiently low level.

Regarding (i), "no radiological consequences" can be interpreted as the absence of non-stochastic effects together with a low individual probability of stochastic effects and a low probability of incurring even one case of stochastic effect in the exposed population.

Siting considerations regarding the "collective dose" involve the comparison of alternative sites in which the favoured ones are those with smallest collective doses. As usually only the local component of the collective dose that varies with sites, the assessment of this component is sufficient in siting and therefore is the rationale behind "population distribution" concepts used in siting.

A rough assessment of the conditions needed to achieve a situation of "no radiological consequences" can be obtained from ICRP Publication No. 26, "Recommendations of the ICRP" [4] and IAEA Safety Series No. 9.

According to these references, non-stochastic effects to individuals are prevented if the annual dose equivalent to any organ does not exceed 0.05 Sv, even if repeated year after year. Therefore, 0.05 Sv in one event cannot cause non-stochastic effects.

The risk of stochastic effects is in the order of 10^{-2} per Sv of effective dose equivalent (ICRP 26 and Safety Series No. 9). If all organs receive doses not exceeding 0.05 Sv, the maximum possible effective dose equivalent would be 0.05 Sv, corresponding to a risk of 5×10^{-4} of lethal cancer or serious genetic effect, a value which is quite small.

The concept "no radiological consequences", i.e., low probability of incurring even one case of stochastic effect in the irradiated population, can be related to the collective dose. A value of 1 manSv corresponds to an expectation value of 0.01 case of stochastic effects. The actual number of cases (0, 1, 2 cases) follows a Poisson distribution, and the probability of not incurring even one case is $e^{-0.01} = 99\%$. Therefore if the values of 0.05 Sv of dose in any organ and one or a few manSv of collective dose are the result of a disruptive event, such event can be considered to have "no radiological consequences".

Clearly, additional* effort will not be required for the extreme event resistant design of a research reactor if it can be shown that:

1. During the lifetime of the facility, extreme events exceeding the design values are not likely to occur (probability of occurrence less than 10^{-2} per annum); and

2. Should such an event occur, there would be "no radiological consequences" in the worst hypothetical case, meaning that it:

   (a) would not result in exposure of any individual of the public exceeding a dose equivalent to any individual organs of 0.05 Sv;

   (b) would not result in a collective dose exceeding a few manSv.

For all installations within the scope of this document, a safety report should be prepared justifying the design assumptions used and, in particular, the consequences of the extreme site-related events in terms of possible damage to the facility and its radiological consequences. The reader is referred to Section 3 of IAEA Safety Series No. 35.

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* Additional to the conventional building and other industrial codes.

2. GENERAL SAFETY CRITERIA FOR THE CLASSIFICATION OF RESEARCH REACTORS

In order to adapt safety requirements to the different types of research reactors, a subdivision into three groups can be proposed and general safety criteria established for each group. The proposed criteria provide guidelines for identifying and designing those systems which should remain functional after an external event in order to meet the requirements of Section 1.2. The general goal to be achieved is to limit the risk of radiation exposure to the public, experimental staff and reactor operators in the event of an accident.

The important factors to consider when grouping research reactors are:

- Excess reactivity;
- Power level and heat removal capability;
- Type of fuel element;
- Fission product inventory;
- Type of containment or of confinement; and
- Type of experimental devices.

2.1. Group I

2.1.1. Classification criteria

The general criterion for including a research reactor in group I is that the conditions of Section 1.2 of the previous chapter, are not violated even in case of collapse of the building, exposure of core or spent fuel to the air by loss of normal leak tightness of the pool or other containment structures and large disruption of the core fuel.

For reactors belonging to this group, the building structure and other components may be designed in accordance with conventional building and other industrial codes.

2.1.2. Safety features

Research reactors can be assigned to this group if, for their characteristics (low power, low excess reactivity, and small fission product inventory), they present no problem of radioactive release nor hazard to the public even in the case of very extensive damage produced by site-related extreme events, such as the rupture of primary coolant boundary and subsequent rapid loss of coolant or a significant break of the reactor building containment boundary.

Usually, zero power reactors and research reactors with an operating power up to approximately 500 KW belong to this group.

For reactors of this group, in order to satisfy the conditions of Section 1.2 it should be ensured that in all circumstances the reactor can be shut down and maintained in shutdown condition.

2.2. Group II

2.2.1. Classification criteria

The general criterion for including a research reactor in group II is that the conditions of Section 1. 2. are not violated provided that the building will not collapse, the pool or other containment structures will not lose normal leak tightness and no big debris will fall on to the fuel or the core.

2.2.2. Safety features

Reactors of this group, because of their characteristics (moderate power, excess reactivity and fission product inventory), need certain simple design requirements for the building and pond to protect against extreme site-related events.

Usually research reactors of an operating power in the range of approximately 500 KW to 2 MW are included in this group.

Also for reactors of this group, in order to satisfy the requirements of section 1.2, it should be ensured that in all circumstances the reactor can be shut down and maintained in shutdown condition. Moreover, the residual heat removal function, which for this group is normally based on natural circulation, should be ensured for the necessary period of time after the site-related extreme event. In fact, many reactors of this group are designed in such a way that partial reactor fuel cooling is available for a certain period of time after the shutdown (e.g. about one hour). In some cases, depending on the power level and heat capacity of the core, it would be necessary to ensure that the coolant is retained for the necessary period of time above a sufficient core height (e.g., one-half).

2.3. Group III

2.3.1. Classification criteria

The general criterion for including a research reactor in group III is that the conditions of Section 1.2 are not violated provided that the building will not collapse, the pool or other containment structures will not lose normal leak tightness, no big debris will fall on the core and that some additional functions are ensured during and after the extreme site-related event.

The main criteria for identifying the sytems which should not lose their function during and after a site-related extreme event are the following:

1. The reactor should be safely shut down and maintained in a safe shutdown condition (earthquake sensors in the shutdown system should be provided and have redundancy and provision for proof testing);

2. All the systems which are not designed to withstand the extreme event must be considered failed;

3. All the systems needed to ensure that the doses are kept within the authorized limits at the external boundary should remain functional. This includes the integrity of the core with regard to safety shutdown and any cooling requirements;

4. For all these safety systems the hypothesis of a single failure of an active reactor component during and after an extreme site-related event should be made;

5. If damage or failure of experimental or irradiation facilities or any fuel storage facilities in the reactor building poses an

important hazard (i.e., could exceed authorized limits at the external boundary) following an extreme site-related event, this should be included in the accident evaluation and criteria 2 and 3 should be applied.

2.3.2. Safety features

Reactors of this group, because of their characteristics (higher power, excess reactivity and fission product inventory), need more complex protection against site-related extreme events.

Research reactors of an operating power above 2 MW and up to a few MW (~5MW) are included in this group. However the general concepts can be extrapolated up to the power of a few tens of MW.

In particular, in order to satisfy the requirements of 1.2, the following functions should be ensured in all circumstances:

- The safe shutdown;
- The residual heat removal; and
- Confinement such as to mitigate the effect of the radioactive releases.

The residual heat removal capabilities to be ensured depend on the results of the thermohydraulic and neutronic calculations for each particular reactor.

For most research reactors, the reactor building is not a leak-tight, pressure containing structure nor does it provide a static fission product barrier. It may be necessary, therefore, to ensure that appropriate ventilation and filtering systems remain functional after an extreme site-related external event.

It may also be necessary to ensure the functions of safety systems which may be associated with experimental facilities.

In Annex II an example is shown on how safety functions have been identified for systems, structures and components for a particular group III research reactor. The table indicates which items should remain functional after an external disturbance in order to prevent a radiological hazard and also to mitigate the consequences of a potential accident.

The identification of important safety functions depends on the type of reactor and the table can only provide guidance for selecting a procedure.

To prevent accidents or mitigate their consequences, it should be ensured that specific safety systems do not lose function.

3. GENERAL SAFETY CRITERIA FOR DESIGN AGAINST EXTREME EXTERNAL EVENTS

Research reactor sites do not always have the same constraints as power reactor sites in terms of water availability, network connection, foundation bearing capacity etc. and can thus be located in areas where the effect of extreme natural and man-induced events is minimized, consequently reducing cost and risk. However, in some cases (for reasons such as the need to be close to research centres or because of the fact that extreme events may affect a large area) this cannot be done and the effects of the site on the installation have to be taken in account, e.g., the external natural phenomena and the external man-induced events which may occur in the site region have to be evaluated, and the reactor has to be protected against the effects of these events.

In the following, geological and geotechnical hazards, earthquakes, flooding and extreme meteorological phenomena will be addressed, as well as man-induced events such as explosions, other industrial hazards and airplane crashes.

The methodologies are simplified ones which can be used for research reactors of groups I, II and III, while maintaining an appropriately high safety level. In cases where the research reactor has a potential for releases of radioactivity not too different from that of a nuclear power plant due to its power level and confinement, it is recommended that the site analysis follow the methodologies indicated in the IAEA NUSS Safety Guides in nuclear power plant siting.

3.1. Extreme Natural Events

Protection against extreme natural events is based on past experiences of these events, which can sometimes be limited. As every extreme natural event yields new findings and modifies the approach to prevention (this is particularly true for those extreme events which occur at wide intervals, such as a big earthquake or volcanic eruption) engineering judgment becomes necessary in considering these unexpected events to avoid imposing too conservative a design basis.

Simplified conservative approaches to evaluate the protection of research reactors against these events are presented below. More sophisticated evaluation to establish the design basis may be performed. In all cases the methods should be reviewed and approved by the Regulatory Body to ensure that they are sufficiently conservative.

3.1.1. Geological and geotechnical hazards*

Geological and geotechnical investigations at the site are performed with the following objectives:

1. To assess possible geological/geotechnical problems involving surface rupture due to faulting, liquefaction, collapse and slope instability;

2. To evaluate the soil characteristics so that a reasonable soil categorization can be achieved (see 3.1.2.2 and Table 3.2);

3. To evaluate geotechnical parameters to be used in the design of the foundation.

The amount of geotechnical investigations to be performed should be based on the extent of potential problems, the available data and the type and size of the facility. For a more comprehensive treatment of the subject the reader is referred to NUSS Safety Series No. 50-SG-S1 "Earthquakes and Associated Topics in Relation to Nuclear Power Plant Siting" [5]; Safety Series No. 50-SG-S2 "Seismic Analysis and Testing of Nuclear Power Plants [6]; and Safety Series No. 50-SG-S8 "Safety Aspects of the Foundations of Nuclear Power Plants" [7].

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* The information in Sections 3.1.1 and 3.1.2 are taken from TEC-DOC 348 and are presented here for completeness. They can be applied to research reactor having a power of up to a few MW. However the methods could be extrapolated to higher power (a few tens of MW) with the approval of the Regulatory Body.

Geotechnical investigations should be carried out to evaluate the bearing capacity and other soil parameters (including hydrogeological conditions) for foundation and building design. These will primarily involve borehole drillings in sufficient number and to sufficient depth, depending on the soil conditions. However, drilling may not be necessary for competent rock sites where the rock formation continues to sufficient depth.

It is recommended that the soil profile be physically identified (e.g. through drilling) to a depth equal to at least one-half of the maximum foundation dimension. The depth to firm bearing strata should also be determined using geological inference, boreholes or geophysical methods.

In parallel with the foundation investigations and the use of available geological/geotechnical data, studies should be performed at the site to assess possible hazards which could result in permanent soil deformation (including surface rupture, liquefaction, collapse, slope instability).

If these investigations indicate particular problems (taking into consideration the load resulting from the design basis earthquake) further studies should be conducted or the site should be rejected.

3.1.2. Earthquakes

For earthquake evaluation and protection against earthquakes it is suggested to refer to IAEA TEC-DOC 348, some information from which is given in subsections 3.1.2.1 to 3.1.2.4 to assist the reader. In applying TEC-DOC 348 it will be noted that, in principle, that group I discussed in the current document corresponds to Class C in the referred document, group II to Class B, and group III to Class A.

3.1.2.1. Design basis earthquake

The design basis earthquake is evaluated on the basis of maximum historical intensity. For this evaluation, the following procedure is applied:

1. A zone having a radius of 100 to 200 kilometres from the site[1] (the higher value is applicable to areas with low seismicity) is selected;
2. Using available publications and catalogues, the maximum observed intensity in the area is established. The information should cover as much historical data as possible, extending to at least 100 years[2];
3. The design intensity level is selected using Table 3.1.

Table 3.1

Range of Maximum Historical Intensity, I_{max} [3]	Design Intensity Level
VII $< I_{max} \leq$ VIII	1
VIII $< I_{max} \leq$ IX	2
IX $< I_{max}$	3

1 If the site is in a relatively quiet seismic zone and the historical seismicity is mainly due to neighbouring active zones, the area for which historical seismicity data is collected could be appropriately adjusted by considering the shape and dimensions of the particular seismotectonic setting.

2 In case of lack of sufficiently old historical data but when at least 70 years of data are available, the maximum historical intensity can be extrapolated over the relevant period, by means of a conservative statistical technique; in other cases the maximum design intensity level can be used or the procedure outlined in Safety Series 50-SG-S1 adopted .

3 Modified Mercalli intensity scale. Even in the region where the maximum historical intensity I_{max} is lower than VII, it is recommended to take the value of VII for reactors of group III.

3.1.2.2. Design basis ground motion

The design acceleration values, a_b for firm bearing strata, corresponding to the design intensity levels of Table 3.1 are given in Table 3.2.

Table 3.2

Design Intensity Level	Design Acceleration for Firm Bearing Strata a_b , in g's *
1	0,08
2	0,15
3	0,30

The acceleration values in Table 3.2 are applied at the firm bearing strata. The amplification effect of geological formations overlying the firm bearing strata is considered in terms of the coefficient γ as $a_g = \gamma\ a_b$, where a_g is the design ground acceleration and the value of γ is given in terms of the soil categorization indicated in Table 3.3.

Table 3.3

Soil Categorization	Categ. 1	Categ.2	Categ.3
Description	Firm Bearing Strata	Other soil than those defined in Categories 1 and 3	Fill ground or Alluvium ground which is thicker than 25 m
γ	1.0	1.25	1.50

A more detailed evaluation may be performed to establish the design group acceleration. However, the method should be reviewed and approved by the Regulatory Body.

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* The acceleration values given do not represent the values corresponding to the upper limit of the intensity range but reference values for the design.

3.1.2.3. Building and equipment design consideration

The seismic design of buildings and equipment can be performed according to the classification of Chapter 2:

Group I: Existing aseismic design code can be applied*;

Group II: Equivalent static approach may be used;

Group III: Simplified dynamic approach should be used.

For explanations on equivalent static approach and simplified dynamic approach, reference should be made to IAEA TEC-DOC 348.

3.1.3. Flooding

Research reactors generally do not need large amounts of cooling water. Therefore, it is not important for them to be located close to a large body of water such as the sea, a lake or a river. It is often possible to select "dry sites", that is, sites which are well above flood level at all times, in both cases of river or coastal sites. For river sites, floods from causes other than precipitation and release from storage should be considered, e.g. mud flows induced by volcanic eruption.

Sometimes, it may not be possible to select "dry sites" simply by inspection. In such cases it is necessary to construct all safety-related items at an altitude above the reference level of the flood, which should be evaluated using methods given below.

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* If there exists no aseismic design code in the country and the maximum historical intensity, I_{max} is lower than VII in Table 3.1, a normal building code which may not include specific aseismic design can be applied.
If there is no aseismic code in the country of design, intensity level 1, 2 or 3 of some other aseismic design code can be referred to. In this case due consideration should be given to the similarity of seismic activity. (Reference: World List of Earthquake Resistance Regulations, International Association of Earthquake Engineering (IAEE), 1984. [8])

3.1.3.1. River sites

For sites located near rivers the reference flood can be evaluated in several ways, e.g.:

- Empirical formulas which have been developed for various parts of the world give a relationship between drainage basin parameters and flood;

- Extrapolated frequency curves, based on a series of maximum annual flows, can be used for evaluating the reference flood. These frequency curves can be derived from available data and taking into consideration random components, trends and jumps. If properly used, this method may allow a reasonable evaluation of a reference flood.

The result of the evaluation should not be less than any recorded historical occurrence.

Having determined a reference flood flow, a reference level can be obtained with appropriate hydraulic formulas which take into consideration the average river channel slope, channel cross section, friction factors etc.

Due consideration should be given to the presence of river channel obstructions close downstream from the site, since they can give origin to backwater elevation at the site.

The effect of dam failures upstream from the site should be evaluated by assuming failure of all dams on the same branch which could influence the site in such a way as to produce the maximum flood at the site.

The wave height evaluated as a consequence of dam failures upstream is transferred downstream at the site section with no attenuation and is superimposed to an appropriate flood level due to precipitation.

Having determined roughly a maximum river flood level, a suitable margin over this level should be assumed, taking into consideration the topography of the site area. This will give the minimum elevation for a dry site.

If it is necessary to use a more sophisticated method then the NUSS Safety Guide SG-50-S10A "Design Basis Flood for Nuclear Power Plants on River Sites" [9] should be referred to.

3.1.3.2. Coastal sites

Also for coastal sites, the best protection is to use a dry site. To establish the reference level for such a site, the potential for coastal flooding should be first evaluated.

If the region of the site is subject to tropical storm effects or if there is a history of tsunamis or of seiches in the area, then historical data on these phenomena must be collected.

The general topography and bathymetry of the site region must be analysed for possible locations which have no possibility of being reached by waters in case of storm surges, tsunamis or seiches.

An analysis of the data available can give a good indication of the maximum flood level at the site. An adequate margin may provide the minimum level for a dry site.

If it is necessary to use a more sophisticated method, then the NUSS Safety Guides 50-SG-S10A and 50-SG-S10B "Design Basis Flood for Nuclear Power Plants on Coastal Sites" [10] should be referred to.

3.1.4. Extreme meteorological events

3.1.4.1. Winds

For installations falling within group I design building code requirements may be used. For installations of groups II and III, design basis winds and cyclones are evaluated on the basis of the maximum historical intensity within a radius of about 100 km from the site. In case of lack of sufficiently old historical data, the maximum historical intensity can be extrapolated over the relevant period by means of a statistical technique, due regard being given to the physical limits of the variable that can be experienced in the area of interest.

For groups II and III facilities, simplified static approach including shape and height effect can be applied for design basis winds; structural members should remain within elastic limit. Consideration should be made of local wind pressure which may cause the failure of external walls.

For reactors of these groups relative elevation over the surrounding areas and effect due to surface roughness, e.g. for locations such as hilltops or seashores, have to be considered in addition to the height of the facilities themselves.

If more sophisticated investigations and analysis are necessary, NUSS Safety Guides 50-SG-S11A, "Extreme Meteorological Events in Nuclear Power Plant Siting, Excluding Tropical Cyclones" [11] and 50-SG-S11B "Design Basis Tropical Cyclone for Nuclear Power Plants" [12] can be referred to.

3.1.4.2. Tornado

For all reactors in group I design building code requirements may be used. For groups II and III approaches similar to those used to evaluate the design basis wind or design basis earthquake can be applied to evaluate the design basis tornado. This means that the plant should be designed for the maximum historical tornado which occurred within a radius of about 100 km from the site. Particular consideration is necessary of the following two phenomena, especially for group II and group III facilities:

- The sudden pressure drop which accompanies the passage of the center of a tornado; and

- The impact of tornado-generated missiles on structures and equipment of the facility.

For the design basis tornado for nuclear power plants it has been suggested to use very heavy missile impacts. However the critical target areas are significantly smaller than those of nuclear power plants, thus the probability of a hit by tornado generated missiles is smaller. Consequently, lighter missiles may be a more realistic design basis for research reactors*.

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* For example, a steel pipe 335 kg in weight, 30 cm in diameter and 4.5 m in length. (USAEC WASH-1361 "Safety-Related Site Parameters of Nuclear Power Plants", 1975 [13]).

3.1.4.3. Snowfall

For group I reactors, design building code requirements may be used. For groups II and III in regions where snow may represent a significant load in the design of plant structures, a design basis snowfall should be determined. Generally, the "expected extreme snowfall" for the reactor's lifetime is used.

It should be noted that in cold regions where snow on the ground may persist for long periods, caution should be exercised in estimating the design basis snow fall since snow compaction varies from place to place.

The meteorological station selected should be one that has a comparable topographical position to that of the proposed site. In mountainous regions where the density of a meteorological network is such that the values measured at the station may be significantly different from those at the site, a site-specific evaluation may be necessary.

The problem of meteorological conditions associated with snowfall, such as wind-induced drifts and blizzard conditions, as well as the problem of avalanches cannot be treated in a statistical manner. They should be evaluated on a case-to-case basis, taking into account any local factors.

Detailed investigations or analyses can be found in NUSS Safety Guide SG-S11A.

3.2. Man-Induced Events

The potential sources of man-induced events external to the research reactor installations should be identified; they should be assessed to determine the potential effect on personnel and safety systems.

The sources of man-induced events can be classified as:

- Stationary, such as chemical plants, oil refineries, oil storage facilities, pipelines;

- Mobile: such as means of transport (road, rail, water, etc.).

The events which may be generated by these sources can be explosions or fire as a consequence of releases of explosives or flammable fluids from stationary or mobile sources, impact from airplane crashes, etc. Usually the consequences of explosions and airplane crashes are of major concern. It is important to remark that the main difference between man-induced events and natural events is that the former can impart a substantial amount of energy to the reactor installation, and this may lead to a possible mechanically induced dispersion of radioactive material. It is thus necessary that research reactors of any of the groups either be located where these events do not affect them in any significant way or they should be protected against them.

3.2.1. Explosions

The site should be located in areas where the effects from explosions are not significant.

The distance from the source of explosion can be evaluated with the methods given in NUSS Safety Series No 50-SG-S5 "External Man-Induced Events in Relation to Nuclear Power Plant Siting" [14] deterministically or probabilistically for fixed sources of explosion, for mobile sources of explosions or for sources of hazardous cloud*. If it is not possible to locate the plant in an area where the risk is not significant, the plant should be protected against these events. Some Member States have adopted a probability limit of 10^{-6} of 10^{-7} per year. In this case, it is necessary to design the building and, sometimes, systems and components against an explosion. Often, the level of the pressure wave which could be created by an event in a fixed installation possibly present around the site is deterministically taken into account. The level of the over pressure for an explosion created by transport should also be considered if this probability is significantly high.

3.2.2. Aircraft crashes

The site should be located in an area where the risk of air crash is not significant.

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* In this Safety Guide it is pointed out that in some Member States a screening distance value of 8-10 km is used for source of hazardous clouds and 5-10 km for sources of explosions.

In agreement with the basic principles discussed in Safety Series 50-SG-S5 and Annex I, two approaches are possible:

- The safe distance from the airport can be evaluated with the adequate formulas given in the above guide*; or

- A probabilistic approach can be used (some Member States have adopted a probability limit of 10^{-6} or 10^{-7} per year). The probability of an aircraft hitting sensitive parts of an installation is related to the size of the installation. In comparison with power reactors, research reactors are limited in size; therefore, for the same events of an aircraft crash, the probability of collision is lower than that for a larger power reactor. The resulting risk, all things being equal, is much smaller.

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* In a footnote and appendix of Safety Guide 50-SG-S5, which present criteria for nuclear power plants, it is pointed out that in some Member States a screening distance value of 10 km is used for all but the biggest airports; among these, there are taken into account airports with projected operation greater than 500 d^2 movement/year located within 16 km of the site and greater than 1000 d^2beyond 16 km (where d is the distance in km from the site).

4. ACCIDENT CONSIDERATIONS AND EMERGENCY PLANNING

The population around every research reactor has to be protected adequately. To achieve this, evaluations should be made of the adequacy of plant safety features for the given population distribution and of the emergency planning feasibility.

It should, however, be considered that accident considerations as presented here allow only a definition of the measures to be taken into account around the nuclear facility. Emergency planning feasibility allows the analysis of aspects of the population distribution. Considerable experience and judgment is necessary to establish the acceptability as a whole of the population distribution around any nuclear facility.

Additional consideration for emergency planning should be taken when the reactor has to be sited near special buildings (such as hospitals, industrial facilities etc.).

4.1. Accident Considerations

A basic criterion for selecting a site for a research reactor or for deciding an appropriate site-reactor combination is that the risk involved in potential accidents is judged to be acceptable.

There have been two approaches to implement this criterion. In the first approach (deterministic) the general requirement for site selection of a proposed reactor is that doses to individuals of the public even in the case of a postulated accident representative of the very substantial ones possible in the reactor (the "design basis accident") are within the prescribed limits set by the Regulatory Body. These limits are the bases for determining the need for additional engineered safety features. The source term for determining the potential exposure of the population is that corresponding to the "design basis accident", which is site- and reactor-specific.

In the second approach (probabilistic), the probability of different possible accidents is considered in connection with the resulting doses.

A rough estimate of the accident term is, in general, sufficient for calculation at the early stage of the siting process and to evaluate the doses to the individuals and to the population. It should be taken into account that some of these source terms represent releases into the building. However, other effects could substantially reduce these releases, e.g. decay or plating out in the containment. For these reasons, the source term may represent an arbitrary upper limit in many cases.

To determine if a reactor/site combination is acceptable, the following first-cut method may be used:

- A spectrum of representative accidents should be postulated;

- The release from the fuel should be estimated for each postulated accident;

- The absorption in the pool water and the release to the building atmosphere should be evaluated as appropriate;

- The deposition of fission products within the building may be taken into account. For first approximation this can be neglected;

- Filter effect and retention effect of the confinement or containment structures may be considered, if appropriate. For first approximation these engineered safety systems can be neglected in dose calculations. Additional engineered safety systems can be considered if dose resulting from simplified calculations is too high, provided that the system can withstand the site-related extreme events and the calculations are of high reliability.

In the deterministic approach, if the doses derived do not exceed the limits prescribed by the Regulatory Body for the most exposed individual, the plant and site combination acceptability is satisfied. However, if the dose to the most exposed individual is greater than authorised limits, the filter and containment or confinement building mitigating effects may be considered taking into account the restrictions given in Section 1.2.

In the probabilistic approach, in case the Regulatory Body has defined criteria, the doses and their probability of occurrence are compared with said criteria. If the criteria are not met, either mitigating features or reliability should be increased reducing the probability of accident.

A low leakage containment structure can reduce the rate of release to the atmosphere by orders of magnitude. A confinement building will release its contents equivalent to the leakage rate or discharge rate from its cleanup systems. This results in a considerable reduction in activity release due to the filtration and the decay of various fission products. Also to be considered are the reduction factors due to deposition in the building. These may be quite high for iodine depending on such factors as residance time, leakage rate, surface material, vapor concentration and species.

4.2. Emergency Planning Feasibility

Some Member States require an emergency plan for research reactors that is independent of dose limits resulting from deterministically postulated accidents, which may be related to both type of reactor and its power.

In other Member States the decision for emergency planning is related to individual or population doses after certain accident hypotheses. The upper limit for the source term may be derived on a case-to-case basis, in order to decide if an emergency plan has to be established. In addition, if an emergency plan is established, the procedure may be used to define important parameters for the emergency planning itself. One possible approach is to consider these source terms and take into consideration only those engineered safety features indicated by the Regulatory Body. The emergency plan would then be extended to the limit where the doses are lower than the emergency reference levels.

4.2.1. For group I

The inherent safety of the reactors of this group prevent significant exposures of the public in the event of extensive postulated accidents. For reactors of this group, it can be demonstrated that there is no need for off-site emergency planning.

However, local or on-site emergency planning will be required to protect personnel in the facility in case of accidents.

4.2.2. For group II

Fuel melting and any important release of radioactive material should prove to be non-credible for all accidents because of the inherent features of the reactor of this group, including seismic and other external forces (e.g. by assuring that sufficient water will always remain in the core for fuel cooling and, in general, releases from the core are very small). Therefore, emergency planning feasibility is not normally required.

If fuel melting or any significant release of radioactivity is considered possible, the feasibility of an emergency plan near the reactor should be demonstrated. On-site emergency planning to protect personnel in the reactor and possibly in a limited zone around the reactor should be required.

4.2.3. For group III

Since for reactors in this group the potential of fuel damage and fission product release is related to the adequacy of the shut-down heat removal system, the requirement for and extent of an emergency plan feasibility has to be established on a case-to-case basis.

Annex I

REASONS FOR SCOPE RESTRICTION OF THE DOCUMENT

The recommendations presented in this document may not be appropriate for already built reactors for the following reasons :

- The need to apply each of the requirements to already built reactors should be considered on a case-to-case basis by evaluating the safety improvements, related cost and the possible related dose commitment for performing the modifications;

- The methods to be used for demonstrating compliance, for already existing reactors, would not necessarily be the same as those described in this document and related references; e.g. methods could be applied taking into account additional margins, such as reserve of structural resistance in the inelastic field, in the case of existing reactors.

- In the case of existing reactors, the failure modes of the system could be considered because some of them would not necessarily involve unacceptable damage and there would then be no need to require upgrading the design against this event.

Therefore for existing reactors, analyses dependent on the specific reactor and site should be conducted to determine the adequacy of the facility or the need for upgrading.

Annex II

TYPICAL SYSTEMS TO BE EVALUATED IN THE SAFETY ANALYSIS WITH REGARD TO REMAIN FUNCTIONAL FOLLOWING AN EXTERNAL DISTURBANCE TO A GROUP III REACTOR

Scope of the system	Structures, Systems and & Components (S.S.C)	Function	Effect from the Loss of Safety Function
	. Control rod drive mechanisms	Prevention of an excess reactivity insertion to the reactor (uncontrolled rod withdrawal)	Damage to the core
Prevention of Accident	. Fuel Elements and . Core Structures . Upper Shieldings . Reactor Water Tank & Concrete Pool	Preservation of core configuration (Blocking of Coolant Channel) Prevention of Mechanical Core Damage	Damage of fuel elements
	. Tank and Concrete Pool Lining	Preservation of water in the Tank to cover the Reactor Core	Inability to sustain Reactor Pool Water Level for any extreme events
	. Control Rod and its Scram Mechanism	Emergency Shut-Down (Scram)	Cooling requirements to be maintained to avoid core damage
	. Water Dumping System (Reflector, Core Dropping System for a Solid Reactor)	Preservation of a reactor in subcritical state after scram	Unsufficient shutdown margin for special reactor types
	. Pipes and pumps to keep the water level sifficient to cover the reactor core . Emergency pump for primary coolant to maintain flow	Heat removal after a reactor shutdown	Damage of fuel elements
Mitigation from the consequences of accidents	Siphone-Brake valve in the pool water level sustaining system	Anti-Siphon Action	Minimum water level to cover the reactor core for LOCA type of events not maintained to avoid core damage
	Emergency air exhaust system, pool water level sustaining system, reactor confinement, gaseous exhaust chimney, isolation valves of gaseous exhaust system	Reduction of the the release of radioactive materials	Higher radioactive releases
	Control and safety protection systems	Generation of the necessary signals	Unability of reactor control

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LIST OF PARTICIPANTS

Working Group, 3-4 December 1984

Bernard, H.	United States of America
Fauré, J.	France
Riotte, H.G.	Germany, Federal Republic of
Sumita, K.	Japan

IAEA

Byszewski, W.	Scientific Secretary
Giuliani, P.	
Iansiti, E.	
Palabrica, R.	

Consultants Group, 30 September - 4 October 1985

Fauré, J.	France
Watabe, M.	Japan

IAEA

Chrysochoides, N.	Scientific Secretary
Giuliani, P.	
Gürpinar, A.	
Iansiti, E.	

Technical Committee, 18 - 22 November 1985

Cancio, D.	Argentina
Mueck, H.	Austria
Christov, C.	Bulgaria
Fauré, J.	France
Kampf, T.	G.D.R.
Riotte, H.G.	Germany, Federal Republic of
Sumita, K.	Japan
Watabe M.	Japan
Campbell, H.E.	U.K.
Bernard, H. (Chairman)	U.S.A.

IAEA

Byszewski, W.	
Chrysochoides, N.	Scientific Secretary
Iansiti, E.	