IAEA
International Atomic Energy Agency
Atoms for Peace and Development

Application of Graded Approach for Site Evaluation for Advanced Reactors – Overall Point View

Interregional Workshop on Graded Approach for Site Evaluation for SMRs in Haikou, China

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Outline

1. What is grading? Background ideas for grading
2. Application of graded Approach based on IAEA Safety Standards
3. A performance-based general approach for grading of siting activities
4. A PSA-based approach for grading of siting activities
5. Final remarks

What is “grading” ?

A ‘graded’ approach is

“An application of safety requirements that is <u>commensurate</u> with the characteristics of the facilities and activities or the source and with the magnitude and likelihood of the exposures”

(IAEA Safety Glossary)

What is "grading" ?

"Grading" is common sense, based on:

- ✓ Radiological hazard and potential impact on health and environment;
- ✓ Possible consequences in case of failure.

It does not make sense that a small radiological hazard facility requires the same effort in siting activities than a facility with the radiological hazard of a regular NPP

- ✓ If the same effort is spent, we may end up overshooting final safety requirements, with an unnecessarily expensive facility.

Key point: It is to demonstrate that whatever "grading" is used, final safety requirements must be met

- ✓ Those requirements are eventually expressed in terms of acceptable doses to workers and public.

Background ideas for “grading”

1. Grading is an open concept, there is not a single way to grading
   (As far as the overarching safety principles are respected, there is not a single way to grading. There may be several acceptable approaches)

2. Radiological risk goals cannot be graded (annual frequencies of exceeding acceptable doses for workers and the public)

3. Risk has two components:

   (a) The annual frequency of failure of the installation (i.e. reliability)

   (b) The (conditional) probability that workers or the public receive unacceptable doses

4. Component (a) depends on how site investigation leading to design basis parameters is carried out.

Background ideas for “grading”

5. Reduction of reliability may be acceptable, as far as the consequences, point (b), are acceptable

6. Consequences depend on the radiological hazard of the nuclear installation.

   Radiological hazard depends on

   (a) Installation type and installation design

   (b) Site conditions (e.g. dominant winds, groundwater flow)

   (c) What is around the installation (e.g. population centers)

Background ideas for “grading”

Rationale behind a graded approach

Safety goal

Same for all nuclear facilities

For example:

Frequency $< 10^{-6}\ yr^{-1}$ of 1000 man-Sv collective dose in 5 km radius

(~30 TBq I-131 release + population of 10000)

Frequency of exceeding dose limit

=

Frequency of failure

x

Conditional probability of exceeding a release source term threshold, given the failure

x

Conditional probability of exceeding dose, given the release

Frequency of hazard (depends on site)

x

Conditional probability of failure for each severity level of hazard (depends on design)

Depends on the layout of the facility (confinement barriers, leak path factors)

Depends on site conditions (wind regime, atmospheric stability, distances to site limits…)

Meeting the safety goal (dose limit) depends on:

- Source term + Population around site
- Site specific hazard (seismic, flood, wind, aircraft crash, etc.)
- Design of the facility against hazards and confinement barriers
- Conditions at the site (meteorological, hydrological, site boundaries…)

Background ideas for “grading”

Points to be noted:

- ✓ Source terms under a threshold will automatically meet safety goals
- ✓ Systems whose failure will **never** produce a release (source term) above a threshold do not require "nuclear" design
- ✓ Systems whose failure **could** produce a release above the threshold could be conservatively designed for an annual frequency of failure smaller than the target frequency for acceptable doses (e.g. 10^{-6} yr^{-1})
- ✓ Larger frequencies of failure can be accepted for those systems if conditional probability of exceeding the acceptable dose level is small.

Larger frequencies of failure could be possible when:

- Larger hazard frequencies (lower design level earthquake, wind, etc.) are considered
- Less demanding design requirements (relaxation of "nuclear" design) are used

Background ideas for "grading"

Site-Installation interaction takes place:

- ✓ Compliance with radiological risk goals (i.e. ultimate safety goals) cannot be assessed without considering both the site and the installation characteristics

In the design process of regular NPPs:

- ✓ Compliance with radiological risk goals is introduced using surrogate requirements

  e.g., maximum annual frequency of exceedance for the design basis earthquake or the maximum acceptable seismically induced CDF can be considered surrogate requirements

- ✓ Surrogate requirements have been 'tuned', in order to meet the ultimate safety requirements for protecting people and the environment from the harmful effects of ionizing radiation.

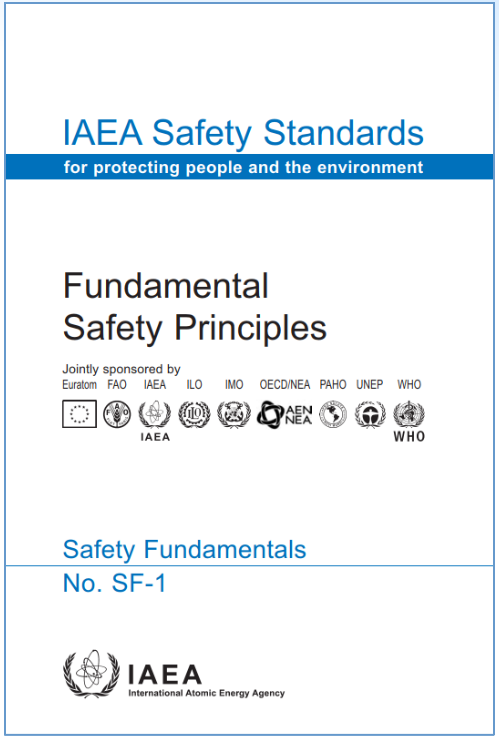
  However, the surrogate requirements for regular NPPs might be overconservative for other, less hazardous, nuclear installations. This is the idea underlying the 'grading' suggested by the IAEA Safety Standards

IAEA Fundamental Safety Principal 5: Optimization of Protection (No. SF-1)

Concept of grading established in IAEA Safety Fundamentals (SF-1), Principle 5

> 3.24. The resources devoted to safety by the licensee, and the scope and stringency of regulations and their application, have to be commensurate with the magnitude of the radiation risks and their amenability to control. Regulatory control may not be needed where this is not warranted by the magnitude of the radiation risks.

… but guidance available in existing IAEA SSGs and supporting publications mainly focused on NIs other NPPs


IAEA Safety Standards
for protecting people and the environment

Fundamental
Safety Principles

Jointly sponsored by
Euratom FAO IAEA ILO IMO OECD/NEA PAHO UNEP WHO

Safety Fundamentals
No. SF-1

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IAEA Fundamental Safety Principal 5: Optimization of Protection (No. SF-1)

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Principle 5: Optimization of protection

Protection must be optimized to provide the highest level of safety that can reasonably be achieved.

3.21. The safety measures that are applied to facilities and activities that give rise to radiation risks are considered optimized if they provide the highest level of safety that can reasonably be achieved throughout the lifetime of the facility or activity, without unduly limiting its utilization.

3.22. To determine whether radiation risks are as low as reasonably achievable, all such risks, whether arising from normal operations or from abnormal or accident conditions, must be assessed (using a graded approach) a priori and periodically reassessed throughout the lifetime of facilities and activities. Where there are interdependences between related actions or between their associated risks (e.g. for different stages of the lifetime of facilities and activities, for risks to different groups or for different steps in radioactive waste management), these must also be considered. Account also has to be taken of uncertainties in knowledge.

3.23. The optimization of protection requires judgements to be made about the relative significance of various factors, including:

— The number of people (workers and the public) who may be exposed to radiation;
— The likelihood of their incurring exposures;
— The magnitude and distribution of radiation doses received;
— Radiation risks arising from foreseeable events;
— Economic, social and environmental factors.

The optimization of protection also means using good practices and common sense to avoid radiation risks as far as is practical in day to day activities.

3.24. The resources devoted to safety by the licensee, and the scope and stringency of regulations and their application, have to be commensurate with the magnitude of the radiation risks and their amenability to control. Regulatory control may not be needed where this is not warranted by the magnitude of the radiation risks.

- Para 3.24 of Principal 5 ‘Optimization of protection’ provided in IAEA SF-1 states that “The resources devoted to safety by the licensee, and the scope and stringency of regulations and their application, have to be commensurate with the magnitude of the radiation risks and their amenability to control. […]”

SITE EVALUATION FOR NUCLEAR INSTALLATIONS, SSR-1

IAEA Safety Standards
for protecting people and the environment

Site Evaluation for Nuclear Installations

Specific Safety Requirements No. SSR-1

IAEA
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4. GENERAL REQUIREMENTS FOR SITE EVALUATION

Requirement 3: Scope of the site evaluation for nuclear installations

The scope of the site evaluation shall encompass factors relating to the site and factors relating to the interaction between the site and the installation, for all operational states and accident conditions, including accidents that could warrant emergency response actions.

4.1. The scope of the site evaluation shall cover all external hazards, monitoring activities and site specific parameters relevant for the safety of the nuclear installation. In determining the scope of the site evaluation, a graded approach shall be applied commensurate with the radiation risk posed to people and the environment.

4.2. The application of the safety requirements for site evaluation for nuclear installations shall be commensurate with the potential hazards associated with the nuclear installation.

4.3. The level of detail needed in the evaluation of a site for a nuclear installation shall be commensurate with the risk associated with the nuclear installation and the site and will differ depending on the type of nuclear installation.

4.4. The scope and level of detail of the site evaluation process necessary to support the safety demonstration for the nuclear installation shall be determined in accordance with a graded approach.

4.5. For site evaluation for nuclear installations other than nuclear power plants, the following shall be taken into consideration in the application of a graded approach:

(a) The amount, type and status of the radioactive inventory at the site (e.g. whether the radioactive material on the site is in solid, liquid and/or gaseous form, and whether the radioactive material is being processed in the nuclear installation or is being stored on the site);
(b) The intrinsic hazards associated with the physical and chemical processes that take place at the nuclear installation;
(c) For research reactors, the thermal power;
(d) The distribution and location of radioactive sources in the nuclear installation;

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SITE EVALUATION FOR NUCLEAR INSTALLATIONS, SSR-1

Requirement 3: Scope of the site evaluation for nuclear installations

- 4.2 The application of the safety requirements for site evaluation for nuclear installations shall be commensurate with the potential hazards associated with the nuclear installation;
- 4.3 The level of detail needed in the evaluation of a site for a nuclear installation shall be commensurate with the risk associated with the nuclear installation and the site and will differ depending on the type of nuclear installation;
- 4.4 The scope and level of detail of the site evaluation process necessary to support the safety demonstration for the nuclear installation shall be determined in accordance with a graded approach.

SITE EVALUATION FOR NUCLEAR INSTALLATIONS, SSR-1

- Requirement 3 of SSR-1 has the objective of ensuring completeness as well as the application of the principle of optimization;
- Completeness check is the first step by a regulator to decide whether or not the Safety Analysis Report can be reviewed;
- The requirement 3 is considered a prerequisite for the completeness of SAR and has to be verified by regulator.
- Scope and level of detail provided in SAR depends on the nuclear installation under review and is determined through the process of grading (wrt. a large NPP).

SITE EVALUATION FOR NUCLEAR INSTALLATIONS, Safety Guides

IAEA Safety Standards
for protecting people and the environment

Seismic Hazards in Site Evaluation for Nuclear Installations

Specific Safety Guide
No. SSG-9 (Rev. 1)

IAEA
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9. EVALUATION OF SEISMIC HAZARDS FOR NUCLEAR INSTALLATIONS OTHER THAN NUCLEAR POWER PLANTS

GENERAL

9.1. The evaluation of seismic hazards for nuclear installations other than nuclear power plants should be commensurate with the complexity of such installations, with the potential radiological hazards and with the hazards due to other materials present on the site.

9.2. The recommended method for applying the graded approach is to start with attributes relating to nuclear power plants and, if possible, to commensurately adjust these for installations with which lesser radiological consequences are associated. If this approach is not practicable for a nuclear installation other than a nuclear power plant, then the recommendations relating to nuclear power plants should be applied.

SCREENING PROCESS

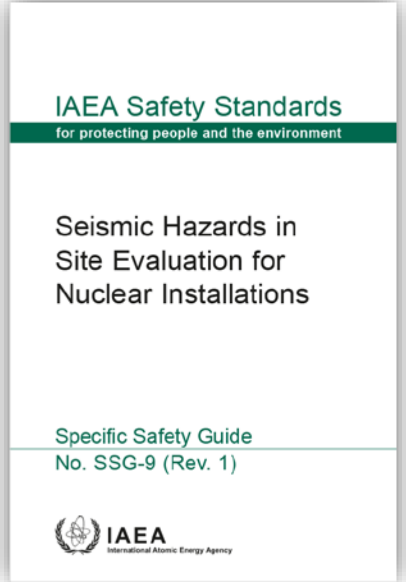
9.3. Prior to adopting a graded approach, a conservative screening process should be applied in which it is assumed that the entire radioactive inventory of the installation is released by the potential seismically initiated accident. If the potential result of such a radioactive release is that unacceptable consequences would not be likely — for workers or the public (i.e. doses to workers and to the public would be below the dose limits established by the regulatory body) or for the environment — and if no other specific requirements are imposed by the regulatory body for such an installation, the installation may be excluded from the requirement to undertake a full seismic hazard assessment. If, even after such a result is reached, some degree of seismic hazard assessment is considered necessary, national seismic codes for hazardous and/or industrial facilities should be used.

9.4. If the results of the conservative screening process show that the potential consequences of such a release would be unacceptable, a seismic hazard assessment of the installation should be carried out, starting from the recommendations relevant to nuclear power plants.

Similar statements can be seen in para 6.5 and 6.6 of SSG-68
&
Para 11.2 of SSG-79

In other words, no account is taken of the safety and mitigation “robust” features embedded in the advanced design

SITE EVALUATION FOR NUCLEAR INSTALLATIONS, Safety Guides



CATEGORIZATION PROCESS

9.9. Three or more categories may be defined on the basis of national practice and criteria, as well as the information described in para. 9.7. As an example, the following categories may be defined:

(a) The lowest hazard category, which includes those nuclear installations for which national building codes for conventional installations (e.g. essential facilities such as hospitals) or for hazardous facilities (e.g. petrochemical or chemical plants) should be applied as a minimum;

(b) The highest hazard category, which includes installations for which standards and codes for nuclear power plants should be applied;

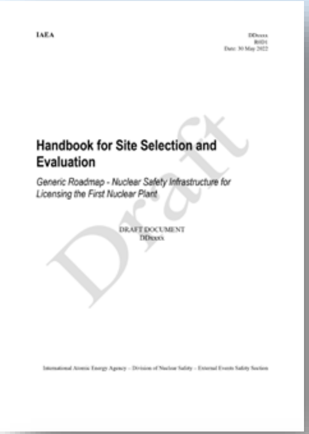
(c) There is often at least one intermediate category between (a) and (b), corresponding to a hazardous installation for which, at a minimum, codes dedicated to hazardous facilities should be applied.

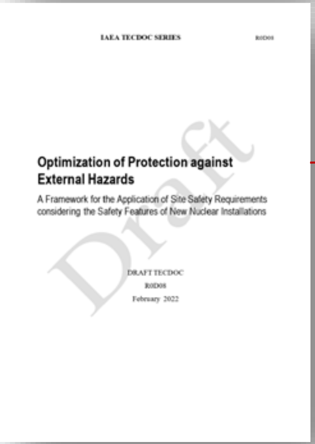
Possible Approaches for Grading of Site Evaluation

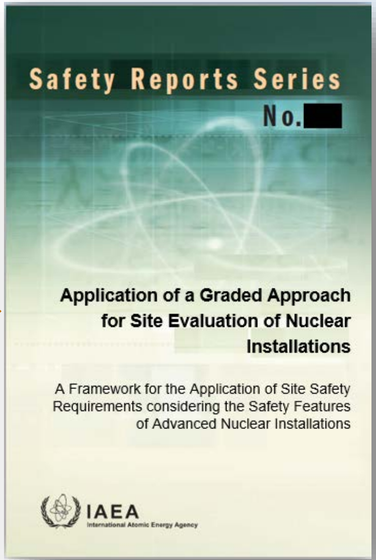
Graded Approach based on:

- ✓ Consequence Analysis (using information/data from safety analysis report developed by the SMRs vendor for a standard design)

- ✓ Probabilistic Safety Assessment (design stage internal/external hazards PSA mostly developed by SMRs vendors for a standard design)

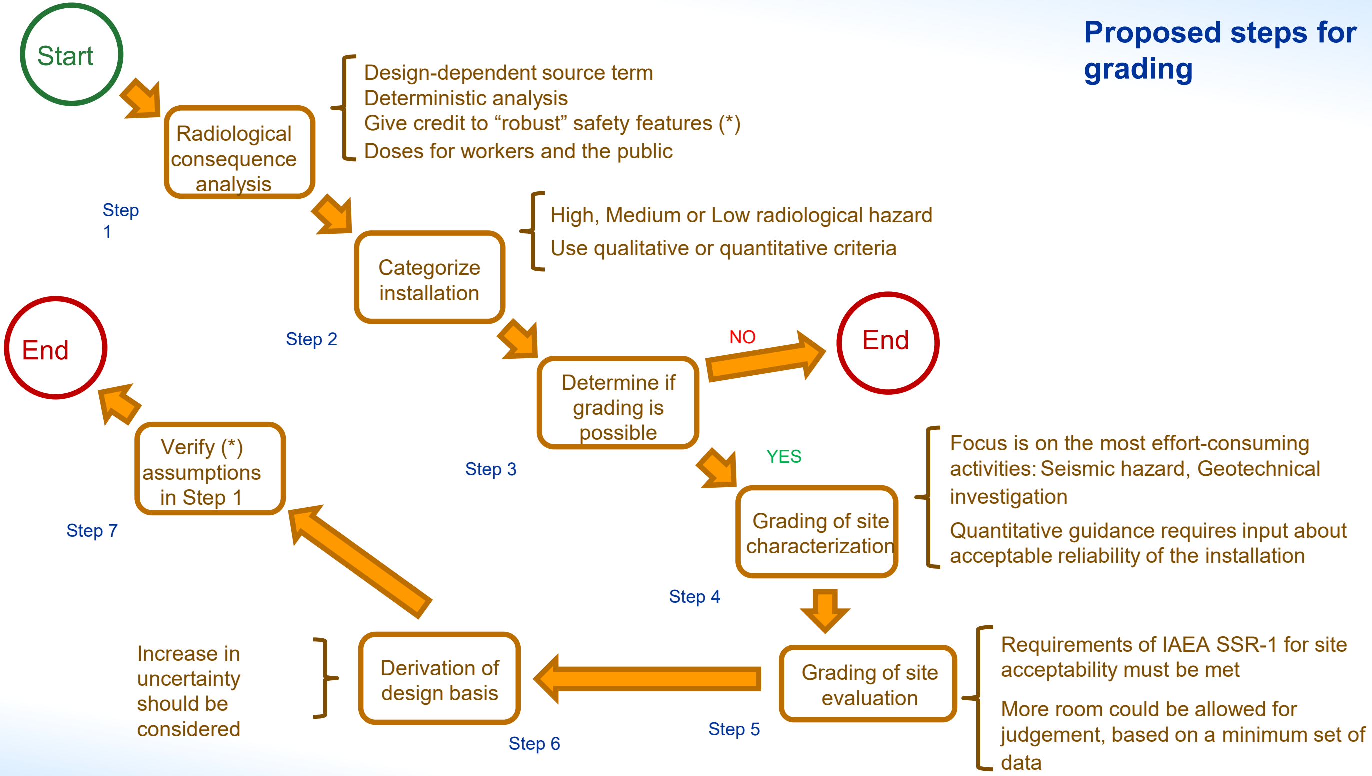






A possible Approach for Grading Site Evaluation-based on Consequence Analysis

Proposed steps for grading



A possible Approach for Grading Site Evaluation-based on Consequence Analysis

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Starting point for grading

First step is to determine the radiological hazard category of the installation…

… by a site-specific radiological consequence analysis

TABLE 1. RADIOLOGICAL HAZARD CATEGORIES BASED ON THE CONSEQUENCES OF FAILURES IN THE NUCLEAR INSTALLATION [10]

Table taken from IAEA SSG-67

Hazard Category	Consequences on site	Consequences off-site	Remarks
High	Radiological or other exposures that might cause loss of life of workers in the facility.	Potential for significant off-site radiological consequences.	Hazard category of nuclear power plants. No grading of site investigation is possible.
Medium	Potential for significant on-site consequences. Unmitigated radiological release would necessitate on-site evacuation.	Small potential for off-site radiological consequences.	
Low	Potential for only localized radiological consequences (within 30–100 m of the point of release).	No radiological consequences.	
Conventional	No radiological consequences.	No radiological consequences.	Site investigation with the same scope as for conventional industrial facilities.

Note: Quantitative definition of 'significant' or 'small' consequences depends on the regulatory framework of each Member State. In the present context, computed doses smaller than 50 mSv are generally considered to be of little consequence.

A possible Approach for Grading Site Evaluation-based on Consequence Analysis

Starting point for grading

The purpose of the consequence analyses is to quantitatively obtain the doses that would be derived from postulated accidents and to compare them with categorization limits

IAEA publications provide only qualitative limits (e.g. IAEA SSG-67)

Quantitative limits accepted by some Member States can be found, for instance in Standard ANS 2.26

A possible Approach for Grading Site Evaluation-based on Consequence Analysis

Starting point for grading

Table A.3 of ANS-2.26-2004

Increasing radiological hazard
(Category 5 corresponds to regular NPPs)

Category	Worker	Public
SDC-1[a)]	No radiological or chemical release consequences but failure of SSCs may place facility workers at risk of physical injury.	No consequences
SDC-2	Lesser radiological or chemical exposures to workers than those in SDC-3 below in this column as well as placing more workers at risk. This corresponds to the criterion in Table 1 that workers will experience no permanent health effects.	Lesser radiological and chemical exposures to the public than those in SDC-3 below in this column, supporting that there are essentially no off-site consequences as stated in Table 1.
SDC-3	0.25 Sv (25 rem) < dose < 1 Sv (100 rem) AEGL2, ERPG2 < concentration < AEGL3, ERPG3. Concentrations may place emergency facility operations at risk, or place several hundred workers at risk.	0.05 Sv (5 rem) < dose < 0.25 Sv (25 rem) AEGL2, ERPG2 < concentration < AEGL3, ERPG 3
SDC-4	1 Sv (100 rem) < dose < 5 Sv (500 rem) concentration > AEGL3, ERPG3	0.25 Sv (25 rem) < dose < 1 Sv (100 rem), > 300 mg sol U intake, concentration > AEGL3, ERPG3
SDC-5	Radiological or toxicological effects may be likely to cause loss of facility worker life.	1 Sv (100 rem) < dose, concentration > AEGL3, ERPG3

[a)] "No radiological or chemical release consequences" or "No consequences" means that material releases that cause health or environment concerns are not expected to occur from failures of SSCs assigned to this category.

A possible Approach for Grading Site Evaluation-based on Consequence Analysis

Starting point for grading

The simplest consequence analysis is based on assuming that all radioactive inventory is released

- ✓ This may be a valid approach for installations with a small inventory (e.g. research reactors) but in the case of SMRs, the approach would lead to the conclusion that no grading is possible.

For SMRs, consequence analyses require definition of:

- ✓ Point of release
- ✓ Radiological source term at the point of release
- ✓ Average meteorological/hydrological conditions
- ✓ Location of site limits

A possible Approach for Grading Site Evaluation-based on Consequence Analysis

Starting point for grading

Postulated Accidents

↓

Radiation Source

↓

Release Characteristics

↓

Atmospheric Diffusion

↓

Radiation Dose

Note the site-installation interaction:

Definition of postulated accidents, source terms and release characteristics require a deep knowledge of the installation.

In the current context, it is the vendor who will provide this information, in the safety analysis report for a standard design.

Atmospheric diffusion and final doses depend on the configuration and meteo/hydro conditions of the site

A possible Approach for Grading Site Evaluation-based on Consequence Analysis

Starting point for grading

For SMRs, a very important input for the consequence analyses is identification of safety features which could be credited for to reduce the final doses

- ✓ Whatever feature is credited, has to show “robustness” to deal with the postulated event (e.g., Para. 6.2.3 and 6.3.2.5 ANS 2.26)
- ✓ It is to be expected that passive safety features (i.e. no change of state is needed) will show “robustness” in most scenarios.

A possible Approach for Grading Site Evaluation-based on Consequence Analysis

Step 3: determine the level of possible grading

From the results of the consequence analyses, the nuclear installation is assigned to a radiological hazard category:

- No grading is possible for installations classified as “high radiological hazard”
  - This corresponds to level 5 in table A.3 of ANS 2.26
  - Same guidelines and procedures as for regular nuclear power plants
- Installations classified as “conventional” are out of our scope (they pose no radiological risk)
  - This corresponds to levels 1 and 2 in table A.3 of ANS 2.26
  - Same procedures and rules as for conventional industrial facilities
- Grading is possible for “medium” and “low” radiological hazard facilities
  - This corresponds to levels 3 and 4 in table A.3 of ANS 2.26
  - New rules need to be derived based on safety performance targets

A possible Approach for Grading Site Evaluation-based on Consequence Analysis

Step 4: grading of site characterization

- Grading could be applied to any of the site investigation and characterization activities

- Interest for grading will normally be in the most time-consuming activities:
  - ✓ Seismic hazard assessment
  - ✓ Geotechnical site investigation

- Guidance for grading the activities requires quantitative input about acceptable reliability of the installation
  - ✓ For instance, guidance on an acceptable level of the SSHAC protocol for seismic hazard assessment, or guidance on about the number of boreholes/tests with respect to a regular NPP at the same site
  - ✓ Acceptable reliability (annual frequency of failure) comes from the radiological hazard category: the less radiological hazard, the less reliability will be acceptable, since doses will be smaller in case of an accident

A possible Approach for Grading Site Evaluation-based on Consequence Analysis

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Step 4: grading of site characterization

- The most difficult part of the approach is to assess how reliability is affected by the grading
  - ✓ This depends on types of nuclear installation. In the present context, ‘reliability’ may be understood as the probability that the nuclear installation meets its fundamental safety functions for a specified period of time.

- Some reference values for acceptable reliability levels depending on the radiological hazard of the nuclear installation (taken from the US nuclear seismic design practice):

ANS 2.26	**SSG-67**	**Relative probability of failure**
SDC-5	High radiological hazard	0.1
SDC-4	Medium radiological hazard	0.4
SDC-3	Low radiological hazard	1.0
SDC-2	Conventional	4.0
SDC-1	Conventional	10.0

x 4   x 10

A possible Approach for Grading Site Evaluation-based on Consequence Analysis

Step 5: grading of site evaluation

Warning: there is a minimum characterization to be performed

- Graded site investigation and characterization must allow for a site safety evaluation according to IAEA SSR-1 requirements
  - ✓ Presence of capable faults
  - ✓ Geotechnical structure of subsurface materials
  - ✓ Static and dynamic geotechnical properties, as necessary to assess stability
  - ✓ Potential presence of complex subsurface conditions (e.g. cavities, expansive soils)
  - ✓ Slope instability
  - ✓ Soil liquefaction
  - ✓ Hydrogeological conditions

    …
- Grading may affect the level of detail, but the scope of the investigation needs always to include these items (including estimates of variability)
  - ✓ More room for judgement, based on a minimum set of data
- Requirements of IAEA SSR-1 for site acceptability must be met

A possible Approach for Grading Site Evaluation-based on Consequence Analysis

Step 6: derivation of design bases

- Grading of site characterization increases (epistemic) uncertainty about site parameters
  - ✓ The larger uncertainty needs to be introduced somehow when deriving design bases. Otherwise, less reliable design bases than expected.
  - ✓ This is a price to be paid due to the grading of characterization
- The design bases parameters determined through graded approach may result in reduced soil strength parameters and increased seismic hazard.
- Reductions in strength or increase in hazard level may be acceptable when large margins are available with the site parameters used in the SMR’s standard plant design
- Grading may result in a less detailed knowledge of the structure of subsurface materials
  - ✓ This needs to be considered by defining a range of variation or just taking the most unfavourable conditions.

A possible Approach for Grading Site Evaluation-based on Consequence Analysis

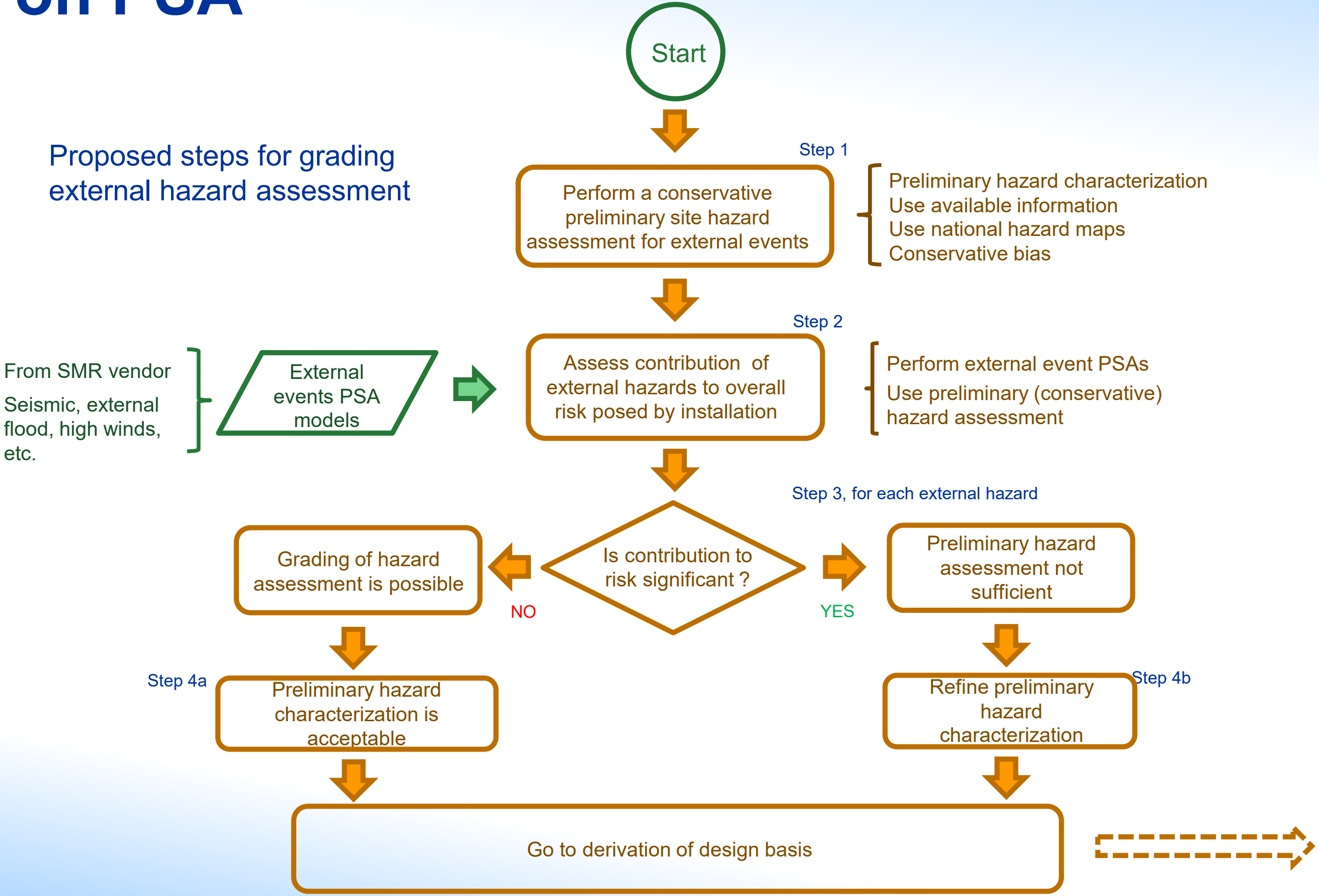
Step 7: Verification of assumptions (confirmation of robustness)

- SSCs credited for mitigation in the consequence analyses need to show robustness to deal with the postulated event
  - ✓ For instance, an SSC credited to keep confinement of radioactive materials after an earthquake, needs to show a large enough seismic margin over the design basis earthquake.

- Confirmation of robustness can be considered as a safety evaluation, after the design is completed
  - ✓ In the case of the earthquake vibratory motion, the verification can be done using the methods described in IAEA SSG-89

A possible Approach for Grading Site Evaluation-based on PSA

Proposed steps for grading external hazard assessment



A possible Approach for Grading Site Evaluation-based on PSA

Site-specific screening of hazards

1. Identification of the hazards and hazard groups that may be possible at the selected site
   - ✓ From a “universe” of potential external hazards, possible hazards at the site are identified. See Section 2 of IAEA TECDOC-1834

2. First screen of potentially safety-relevant hazards
   - ✓ This is done for regular NPPs as well (IAEA SSG-61, Section 3.2)
   - ✓ The screening can be performed in two steps:

     Step A. Based on qualitative criteria, eliminate those hazards that clearly will not have safety consequences or whose consequences are enveloped by other hazards.

     Step B. Bounding analyses, use simple conservative calculations to show that the worst possibility of a particular hazard or a combination of will not have safety consequences. It needs basic characterization of the site obtained, for instance, from the site selection process.

A possible Approach for Grading Site Evaluation-based on PSA

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Site-specific screening of hazards

3. For the screened-in hazards and hazard combinations: develop a preliminary estimate of annual frequencies of occurrence or of exceedance, and associated parameters (e.g. loading, magnitude)
4. For each screened-in hazard, use the external event PSAs to derive applicable risk metrics (CDF, LERF)
5. For each screened-in hazard, assess PSA-derived risk metrics (CDF, LERF) to determine if the hazard could be risk-significant
   - ✓ Thresholds of risk significance for SMR are a subject of discussion
   - ✓ They could be determined from a radiological consequence analyses of postulated accidents (see previous sessions)
   - ✓ For regular NPPs, an external event-induced CDF smaller that 10^{-5} yr-1 is considered acceptable in some Member States
6. Potentially risk significant hazards needs to be refined.
   - ✓ The other external hazards can keep the preliminary hazard assessment, since their contribution to overall risk is small

High Hazards Nuclear Installations

1 Preliminary estimate of the selected hazards and site characteristics

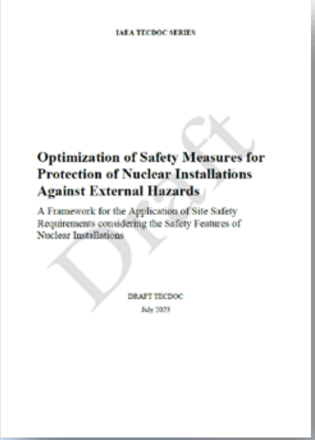
2 Use PSA to identify the critical hazards and dominant risk contributors

3 Develop/refine performance targets and graded approach rules supported by risk ranking

4 Refine hazard and risk assessment for critical contrbutors and perform sensivity analysis

5 Check effectiveness of DiD

Iteration can be stopped when acceptable optimized DiD is reached


IAEA TECDOC SERIES

Optimization of Safety Measures for Protection of Nuclear Installations Against External Hazards

A Framework for the Application of Site Safety Requirements considering the Safety Features of Nuclear Installations

Draft

DRAFT TECDOC

July 2023

General Framework for TECDOC on Optimization of Protection against External Hazards

A possible Approach for Grading Site Evaluation-based on PSA

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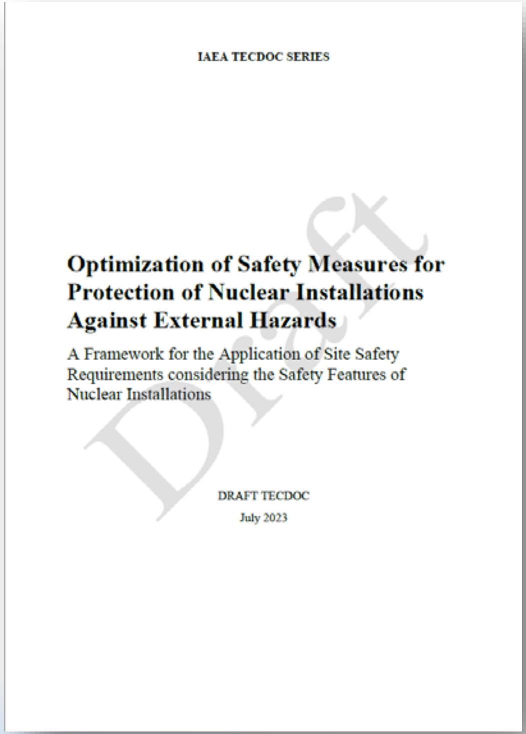
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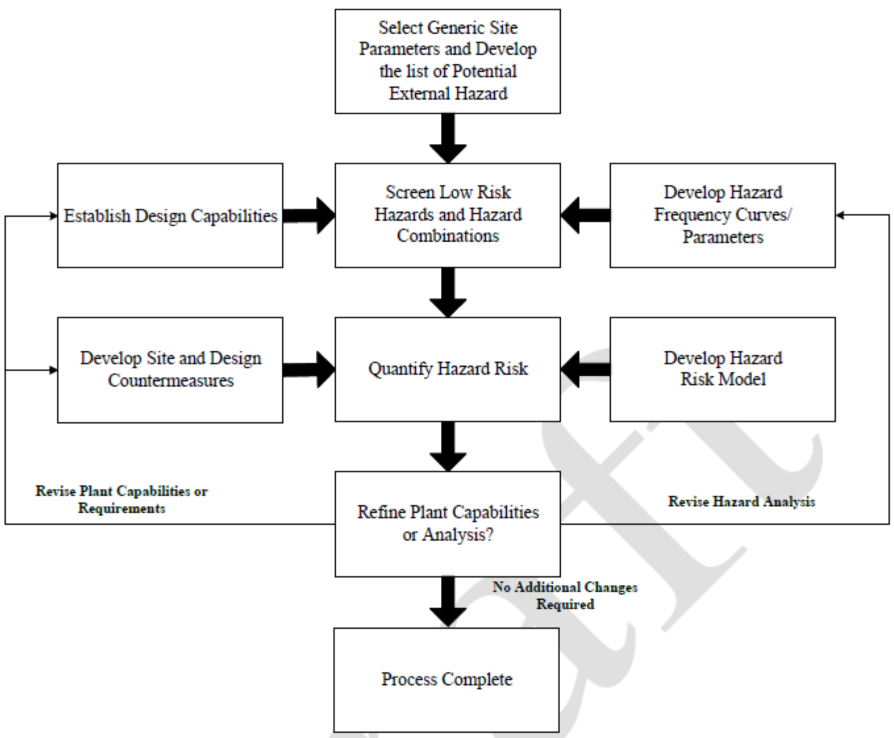

General process of optimization of safety measures in relation to hazards prior to Site Selection

Final Remarks

- Grading is covered in IAEA Safety Standards
  - ✓ It is not a new concept, but application to SMRs with advanced safety features needs to be tailored by developing practical guidance through MSs experience feedback.
- Grading is an open concept: there may be several ways to grade siting activities
- A graded approach might be acceptable as far as it respects final safety requirements
- In a risk-informed performance-based framework, those final safety requirements are given as maximum annual frequencies of exceeding threshold doses to workers and the public
- Following consequence analysis approach grading depends on:
  - ✓ The radiological hazard category of nuclear installation
- A PSA-based graded approach provides the idea to reduce efforts in hazards assessment for those hazards with small contribution to overall risk posed by the plant.
- Grading of site characterization increases (epistemic) uncertainty for site-based design parameters
  - ✓ The larger uncertainty needs to be introduced somehow when deriving design bases

Acknowledgement

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Thank you!

Questions?

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