الهيئة الاتحادية للرقابة النووية
FEDERAL AUTHORITY FOR NUCLEAR REGULATION

Regulatory Guide

EVALUATION CRITERIA FOR PROBABILISTIC SAFETY TARGETS AND DESIGN REQUIREMENTS

FANR-RG-004

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Basic Principle of Regulatory Guides

Regulatory Guides are issued to describe methods and/or criteria acceptable to the Authority for meeting and implementing specific requirements in the Authority's regulations. Regulatory Guides are not substitutes for regulations, and compliance with them is not required. Methods of complying with the requirements in regulations different from the guidance set forth by the regulatory guide can be acceptable if the alternatives provide assurance that the requirements are met.

Definitions

Article (1)

For the purposes of this regulatory guide, the following terms shall have the meanings set forth below.

AC	Alternating Current
Anticipated Operational Occurrence (AOO)	An operational process deviating from normal Operation which is expected to occur at least once during the operating lifetime of a Nuclear Facility but which, in view of appropriate design provisions, does not cause any significant damage to Items Important to Safety or lead to Accident Conditions.
Anticipated Transient without Scram (ATWS)	An anticipated operational occurrence followed by the failure of the reactor protection system.
Beyond Design Basis Accidents (BDBA)	Accident conditions more severe than a Design Basis Accident.
Coolable Core Geometry	Fuel assembly rod bundles retain a geometry with adequate coolant channels to permit removal of residual heat.
Core Damage Frequency (CDF)	The likelihood of Accidents that would cause damage to a reactor core; the sum of the frequencies of those Accidents that result in uncovery and heat-up of the reactor core to the point at which prolonged oxidation and severe Fuel Damage are anticipated and involving enough of the core, resulting into fission products release from the fuel that if released to the environment would result in offsite public health effects.

Defence-in-Depth	A hierarchical deployment of different levels of diverse equipment and procedures to prevent the escalation of Anticipated Operational Occurrences and to maintain the effectiveness of physical barriers placed between a Radiation Source or Radioactive Material and workers, members of the public or the environment, in operational states and, for some barriers, in Accident Conditions.
Design Basis Accident (DBA)	Accident Conditions against which a Nuclear Facility is designed according to established design criteria, and for which the damage to the fuel and the release of Radioactive Material are kept within authorised limits.
ECCS	Emergency Core Cooling System
Fuel Damage	Any fuel relocation, fuel-clad interaction or clad degradation that limits the fuel lifetime, power level or compromises assumptions in the Safety analysis.
Functional Event Sequences (FS)	A group of similar Accident sequences into an event class. Similar Accident sequences are those that have similar initiating events and display similar Accident behaviour in terms of system failures and/or phenomena and lead to similar end states. Similar Accident sequences are likely to have the same systems, structures and components credited for Accident prevention and/or mitigation.
Large Release Frequency (LRF)	The sum of the frequencies of those Accidents leading to unmitigated release of airborne fission products from the Containment to the environment such that there is the potential for health effects. (Such Accidents generally include releases associated with Containment failure, Containment bypass events, or loss of Containment isolation.)
PTS	Pressurised Thermal Shock
Probabilistic Risk Assessment (PRA)	A comprehensive, structured approach to identifying failure scenarios constituting a conceptual and mathematical tool for deriving numerical estimates of risk. Level 1 comprises the Assessment of failures leading to the determination of the frequency of core damage. Level 2 constitutes the Assessment of Containment response and leads to the determination of frequency of Containment failure resulting in release to the environment of a given percentage of the reactor core's inventory of radionuclides.
RCS	Reactor Coolant System
RHR	Residual Heat Removal system

RPS	Reactor Protection System
Severe Accidents	Accident conditions more severe than a DBA and involving significant core degradation.
Structures, Systems and Components (SSCs)	A general term encompassing all the elements of a Facility or Activity which contributes to protection and safety, except human factors. Structures are the passive elements such as building vessels and shielding. A System comprises several components assembled in such a way as to perform a specific active function and a Component is a discrete element of a system.

Background

Article (2)

FANR-REG-03, "Regulation for the Design of Nuclear Power Plants" (Reference 1) contains requirements for the Design of Nuclear Power Plant (NPP) in the State. This regulatory guide contains evaluation criteria which the staff will use to assess the adequacy of the Design, i.e., the performance expected by the NPP and its SSCs with respect to ensuring the protection of public health and Safety and the environment. The evaluation criteria in this regulatory guide are stated, as much as possible, in a performance oriented fashion so as to allow flexibility in the Design, Operation and Maintenance measures used to ensure requirements are met.

Objective

Article (3)

The objective of this regulatory guide is to define evaluation criteria the staff will use in assessing plant Safety assessments associated with probabilistic Safety targets and the Design requirements in FANR-REG-03. The evaluation criteria are intended to be compatible and consistent with requirements in other Authority regulations. If a conflict exists between the evaluation criteria in this regulatory guide and any requirements in other Authority regulations, the requirements in the Authority's regulations are overriding.

Scope

Article (4)

The evaluation criteria contained in this regulatory guide apply to selected requirements in FANR-REG-03 deemed to require further clarification. Accordingly, the articles in this guide identify the FANR-REG-03 articles to which they apply. In some cases this may be multiple articles.

General Requirements

Reliability Goals

Article (5)

(This article applies to Article 4 (item 1) in FANR-REG-03)

Reliability goals necessary for plant Safety should be established. The reliability goals may be established at the system or component level during Design and should be realistic and consistent with assumptions in the PRA (reference 2). These goals should be documented and performance monitored over the life of the NPP, such that a determination can be made as to whether or not the goals are met.

Probabilistic Safety Targets – Evaluation Criteria

Article (6)

(This article applies to Articles 5 and 46 in FANR-REG-03)

1. The NPP should be designed, operated and maintained so as to limit its overall core damage frequency (CDF) to $< 10^{-5}$/yr (mean value from the PRA[1] considering internal and external events and all modes of Operation).
2. The NPP should be designed, operated and maintained so as to limit its overall large release frequency (LRF) to $< 10^{-6}$/yr (mean value from the PRA considering internal and external events and all modes of Operation).
3. The NPP should be designed, operated and maintained so as to avoid a disproportionate concentration of risk resulting from any single SSC failure or human action.
4. Sensitivity studies, using the PRA, should be performed to determine whether small variations in SSC and human performance (e.g., reliability, availability) would cause any of the above evaluation criteria to be exceeded. If the results of the sensitivity studies show any of the above evaluation criteria are exceeded, a review should be conducted and documented to see if corrective action is warranted.

[1] The PRA should meet the requirements in FANR-REG-05 “Regulation for the Application of PRA at Nuclear Facilities” (Reference (2)).

Principal Technical Requirements

Defence-In-Depth

Article (7)

(This article applies to Articles 7, 8, and 13 in FANR-REG-03)

The NPP should be designed, operated and maintained with sufficient Defence-in-Depth (DiD) so as to accomplish the fundamental Safety functions specified in Article 8, Item 1, of Ref (1). Specifically, the Defence-in-Depth approach and evaluation criteria should be consistent with Table 1.

Table 1 – Defence-in-Depth Approach and Associated Evaluation criteria

	Normal Operation	**Anticipated Operational Occurrences**	**Design Basis Events**	**Severe Accident Considerations**
Level of DiD	**1**	**2**	**3**	**4**
Objective	Prevention of abnormal Operation and failure	Control of abnormal Operation and detection of failures	Control of plant Accident conditions, including prevention of core melt and Accident progression, and mitigation of the consequences of Accidents.	Control of Severe Accident conditions, including prevention of Accident progression, and mitigation of the consequences of Severe Accidents.
Essential Features	Conservative Design and quality in Construction and Operation	Control, limiting and protection systems and other surveillance features	Engineered Safety features and Accident procedures	Complementary measures and Accident Management (See Article 12)

	Normal Operation	Anticipated Operational Occurrences	Design Basis Events	Severe Accident Considerations
Reactivity Control	Normal operating activities. Reactivity related variables are kept within a defined safe operating envelope.	a) Challenges should be mitigated by normal operating systems. b) Safe shutdown; and c) Diverse means of accomplishing shutdown function. (See Article 9)	a) Safe shutdown b) At least one means of reactor shutdown is available. (See Article 10) c) Plant is protected such that an ATWS[2] will not lead to conditions that result in a Severe Accident. (See Article 11, item b)	No re-criticality during a Severe Accident
Core Heat Removal	Normal operating activities. Thermal-hydraulic variables are kept within a defined safe operating envelope.	No loss of integrity of reactor coolant pressure boundary or Fuel Damage (See Article 9).	Core heat removal capability is maintained (see Article 11, item a) Coolable Geometry is maintained. (See Article 10) Prevention of reactor pressure vessel failure (see Article 11 item c)	Plant is capable of preventing or mitigating Severe Accidents. (see Article 12)

[2] ATWS is Beyond Design Basis Event and the intent of this guidance is to ensure that it does not progress into a severe accident.

	Normal Operation	**Anticipated Operational Occurrences**	**Design Basis Events**	**Severe Accident Considerations**
Confinement of Radioactive Materials	Normal operating activities.	Normal operating activities (See Article 9).	No loss of function (See Article 10).	Plant includes measures for preventing and mitigating Severe Accidents (see Article 12) Containment integrity is maintained for approximately 24 hrs. following core melt. After 24 hours releases should be controlled (or a conditional containment failure probability of 0.1). (see Article 12, item f)

Requirements for Plant Design

General Design Basis – Normal Operation

Article (8)

(This article applies to Articles 12, 21 and 22 in FANR-REG-03)

1. A set of Operating Limits and Conditions consisting of Design limits for key physical parameters should be defined that document safe, normal operating conditions (i.e., full power, low power, and shutdown) for the NPP. This should include the following parameters:

   a) NPP Design lifetime

   b) Maximum power level

   c) Maximum fuel burn-up

   d) Maximum fuel linear heat rate

e) Maximum core power to flow ratio

f) Primary coolant system Design temperature and pressure

g) Steam and feedwater system Design temperature and pressure

h) Maximum primary coolant system circulating activity

i) Containment Design temperature, pressure and leak rate Fuel pool coolant inventory (e.g., water level) and coolant temperature

2. The annual Effective Dose to the operating staff and to members of the public from normal Operation should not exceed the Dose limits stated in FANR-REG-04 “Regulation for Radiation Dose Limits and Optimisation of Radiation Protection at Nuclear Facilities”, Reference (3).

General Design Basis – Anticipated Operational Occurrences (AOOs)

Article (9)

(This article applies to Articles 12, 21, 22, 48 (Item 2), 50, 51 and 55 in FANR-REG-03)

The following should be used as evaluation criteria in the analysis of AOOs:

1. Fuel (reactor) – no damage or loss of lifetime (i.e., no centreline fuel melting, no departure from nucleate boiling, fuel enthalpy below limits shown in Figure B-1 of Section 4.2 of Reference (4)).

2. Reactivity control – maintain redundant and diverse means for reactor shutdown.

3. Fuel (spent fuel pool) – maintain coolant inventory and $K_{eff} < 0.95$ (at 95% confidence value).

4. Reactor Coolant System – no damage or loss of lifetime (i.e., remain within Design conditions for normal Operation).

5. Residual Heat Removal system – no loss of function.

6. Containment – no damage or loss of integrity (i.e., no degradation in Design leak rate).

7. The annual Effective Dose to members of the public from AOOs should not exceed the Dose limits stated in Article (4) of Reference (3).

General Design Basis – Design Basis Accidents (DBAs)

Article (10)

(This article applies to Articles 12, 21, 22, 48 (Item 3), 50, 51, 56, 57, 58, 59 (Item 1), 62 (Item 1), and 72 (Item 1) in FANR-REG-03)

The following should be used as evaluation criteria in the analysis of DBAs:

1. Fuel (reactor)
   a) maximum clad temperature – 1205°C
   b) maximum fuel enthalpy – 230 calories/gm
   c) maximum clad oxidation – 17 percent
2. Reactivity control
   a) maintain at least one means for reactor shutdown
3. Fuel (spent fuel pool)
   a) maintain active fuel covered in water
   b) maintain $K_{eff} < 1.0$ (at 95% confidence value).
4. Reactor Coolant System (RCS)
   a) maintain RCS integrity
5. Emergency Core Cooling System (ECCS)[3]
   a) maintain Coolable Core Geometry
6. Emergency feedwater system
   a) Maintain Coolable Core Geometry and achieve cold shutdown
7. Residual Heat Removal System (RHR)
   a) Maintain Coolable Core Geometry and cold shutdown.
8. Containment
   a) maintain Design Basis leak rate

[3] The Design of the Containment sump debris screen should not prevent the required amount of water being provided to the core (as a consequence of the accumulation of debris) in the event of a loss of coolant accident as well as to any Safety equipment cooled by the pumps.

b) maintain temperature and pressure below Design limits

9. Control Room

   a) The control room shall remain habitable from the effects of radiation and toxic gas resulting from an Accident, as described in Reference 5.

10. The Effective Dose from DBAs, calculated on the following basis, should not exceed 0.25 Sv:

    a) The Dose should be calculated at the site boundary considering an unsheltered individual located at the site boundary for two hours of release duration or until evacuation can be assured. Assumptions used in the Dose calculation should be consistent with the following:

       - The amount, timing and chemical form of Radioactive Material released into the Containment (i.e., source term) as a result of a DBA should be based upon a mechanistic analysis of the Accident scenario. The source term described in NUREG-1465 (Reference 6) for an in-vessel core melt Accident is acceptable for use in the large break loss of coolant Accident analysis.

       - The Containment should be assumed to leak at its Design basis leak rate for the duration of the Accident.

       - Credit may be taken for attenuation of the in-Containment source term (e.g., Containment sprays) provided the attenuation assumed can be justified.

       - Realistically conservative meteorology (e.g., wind speed, direction) should be assumed so as to envelope the most likely site meteorology.

    b) Alternative methodology and assumptions could be used with adequate justifications.

Additional Design Considerations

Article (11)

(This article applies to Article 24 in FANR-REG-03)

Recent studies using realistic assumptions, state-of-the-art tools and data, and insights from operating experience and research show that some scenarios that have contributed significantly to risk in previous studies have reduced frequencies as compared to earlier estimates due to the incorporation of new Design features and procedures. These include:

1. Loss of All Alternating Current (AC) Power (Station Blackout)

   A station blackout involves the complete loss of AC electric power to the essential and nonessential switchgear buses in the NPP. This involves a loss of offsite power followed by failure of all diesel generators. This situation results in reliance solely on station battery power which, when exhausted, could result in core melt and conditions that could threaten the

Containment integrity. If the PRA shows that a station blackout sequence contributes significantly to risk, a diverse alternate power source should be included in the design, (e.g., gas turbine generator with capability to power one complete set of normal safe shutdown loads for either 24 hours or until safe shutdown can be maintained). The station battery capacity should be sufficient to power critical loads such that reactor core cooling is maintained until such time as the alternate power source can be brought on line. In addition, unless the integrity of reactor coolant pump seals under station blackout conditions can be demonstrated, the Design should also include a backup seal injection pump powered by a small dedicated diesel generator, which has enough capacity to also charge the station batteries.

2. Anticipated Transients Without Scram (ATWS)

An ATWS is an Anticipated Operational Occurrence followed by the failure of the reactor protection system (RPS). The transients, coupled with a failure of RPS, may lead to conditions beyond the Design Safety limits and absent prompt action to shut down the reactor could lead to Fuel Damage and/or coolant system damage. For example, loss of feed water transients followed by failure to shut down the reactor has the potential for damage to the reactor core, since these scenarios lead to a significant increase in reactor pressure. In order to protect against a common mode failure that could lead to failure of the RPS, diverse means (e.g., sensors, actuation devices) should be provided to assure that reactor power, pressure, and temperature are controlled. The applicant/Licensee should either demonstrate that the consequences of an ATWS are acceptable (e.g., fuel integrity is maintained), or provide Design features for diverse and independent means of shutting the reactor down. If independence and diversity is only provided for the actuation portion (excluding the rods) of the reactor shutdown system, other measures may be necessary to demonstrate acceptable consequences, such as the diverse actuation of turbine trip and initiation of Emergency feed water.

3. Fracture Toughness Requirements for Protection Against Pressurised Thermal Shock (PTS) Events

PTS means an event or transient where a severe overcooling (thermal shock) of the reactor pressure vessel occurs concurrent with or followed by moderate pressure in the reactor vessel. Failure to adequately prevent PTS events could result in reactor vessel failure, core melt, and conditions that could threaten Containment integrity. Embrittlement of steels, particularly reactor pressure vessel steel and its welds, due to exposure to fast neutrons can reduce the ability of the reactor vessel to withstand thermal and mechanical stresses. In order to protect against PTS, reactor pressure vessel materials that resist embrittlement should be used and beltline welds should be minimized or eliminated. In addition, Design features that reduce the neutron flux incident on the reactor vessel may also be used. The level of embrittlement of reactor vessel materials should be monitored, and supported by a reactor vessel material surveillance programme. Properties of the irradiated reactor vessel materials samples should be measured periodically, the amount of embrittlement determined and the effectiveness of any corrective actions should be evaluated. Reference 7 on fracture toughness requirements for protection against pressurised thermal shock events provides additional information and evaluation criteria.

4. Protection During Refueling and Maintenance (Mid-Loop Operation)

During PWR refueling or maintenance (e.g., steam generator) activities, the reactor coolant system (RCS) water inventory is sometimes reduced to a “mid-loop” level with fuel in the core. The reduced reactor coolant inventory at mid-loop provides less time to recover in the event of off normal events. For example, during this period, the potential exists for loss of decay heat removal capability or additional water inventory. The design should incorporate capability to monitor reactor vessel water level and temperature, ensure high reliability of the shutdown decay heat removal system and capability to quickly close the containment during this plant configuration. In addition, controls should be put in place to limit the time at which the facility is in this configuration.

Severe Accident Considerations

Article (12)

(This article applies to Articles 12, 24, 59 (Item 2), 60 (Item 2), 67 (Item 2), and 68 (Item 2) in FANR-REG-03)

1. Because it takes time for an Accident to progress and the transport of radionuclides into the Containment is gradual and does not include the entire inventory due to deposition on colder surfaces in primary and secondary systems, and because of a better estimate of Containment performance under Severe Accident loads, releases to the environment and subsequent consequences are significantly reduced.

2. Accordingly, the Severe Accident considerations discussed in this article capitalise on the results of Severe Accident research that was conducted over the last 30 years. There is now a greater understanding of what happens during a Severe Accident, which allows a better estimate of Containment performance under Severe Accident loads, and more reliable Severe Accident management and Emergency Preparedness programmes.

3. The Severe Accident considerations discussed in this article are intended to ensure that the likelihood of an Accident having harmful consequences remains extremely low, i.e., reduce to low likelihood the probability of occurrence of core melt Accident and/or acute radiation exposures resulting in fatalities. The incorporation of such features provides Defence-in-Depth and helps compensate for phenomenological and other uncertainties (e.g., human error) that affect the risk from Severe Accidents. Designs meeting the evaluation criteria discussed below can be considered to have effective Severe Accident prevention and mitigation capabilities and provide adequate assurance of protecting public health and Safety.

   a) In-vessel Core Melt Retention

   During a core melt Accident, if the reactor vessel remains intact, molten core debris will be retained in the lower head and phenomena such as ex-vessel steam explosion, direct Containment heating, and core concrete interactions, which occur as a result of core debris relocation to the reactor cavity, can be prevented. Measures to promote in-vessel retention, such as by flooding the reactor cavity with water to cool the core debris inside the reactor pressure vessel may be included for use as an Accident management strategy. However, with the low frequency of core melt Accidents specified in Article (6), and the Severe Accident mitigation features listed in items (e) through (j) below, additional in-vessel retention measures would not be warranted unless the PRA shows this to be a key feature for the protection of public health and safety.

b) Steam Explosions

- In-Vessel Steam Explosion

  During the initial stages of progression of Severe Accidents, molten debris from the damaged core would relocate to the lower plenum of the reactor pressure vessel. If a sufficient amount of water remained in the lower plenum, the molten core material falling into the water could generate steam and if severe enough, an explosion. This explosion could challenge the reactor vessel and Containment integrity. However, a recent assessment of this issue by a United States Nuclear Regulatory Commission sponsored steam explosion review group (Reference (8)) concluded that this mode of Containment failure has a very low likelihood of occurring. It should be confirmed that the underlying assumptions in Reference (8) are applicable to the proposed Design.

- Ex-Vessel Steam Explosion

  Reactor vessel failure at high or low pressure coincident with water present within the reactor cavity may lead to interactions between fuel and coolant with a potential for steam generation or steam explosions. Steam explosions involve the rapid mixing of finely fragmented core debris with surrounding water resulting in rapid vaporization and acceleration of surrounding water creating substantial pressure and impact loads. It should be confirmed that the Design has been analysed for ex-vessel steam explosion and that the structural integrity of the Containment would be maintained in the event of an ex-vessel steam explosion.

c) Combustible Gas Generation and Control

The issue regarding combustible gas generation centres on the rate and quantity of hydrogen production and the associated hydrogen steam mass and energy release rates into the Containment during both in-vessel and ex-vessel phases of Severe Accidents. These parameters strongly influence the flammability of the Containment atmosphere and the magnitude, timing, and location of potential hydrogen combustion. Hydrogen combustion in the Containment could produce pressure and thermal loads that might threaten the integrity of the Containment boundary. There are uncertainties in the phenomenological knowledge of hydrogen generation and combustion. In order to ensure Containment integrity will be maintained, the Design should provide a system for hydrogen control that can safely accommodate hydrogen generated by the equivalent of a 100 percent fuel-clad metal-water reaction. In addition, the Design should be capable of precluding uniform concentrations of hydrogen from exceeding 10 percent (by volume), or should provide an inerted atmosphere within the Containment.

d) Core Debris-Concrete Interaction

In the event of a Severe Accident in which the core has melted through the reactor vessel, it is possible that Containment integrity could be breached if the molten core is not sufficiently cooled. In addition, interactions between the core debris and concrete could generate large quantities of additional hydrogen and other non-condensable gases, which could contribute to the eventual overpressure failure of the Containment. Downward erosion of the basemat concrete could also lead to basemat penetration with the potential for ground water contamination and subsequent discharge of radionuclides to the surface environment. Thermal attack by molten corium on retaining sidewalls could produce structural failure within the Containment causing damage to vital systems and perhaps to failure of Containment boundary. Therefore, the applicant/licensee should assess a) reactor cavity floor space to ensure adequate area for debris spreading; b)means to flood the reactor cavity to assist in the cooling process; and c) impact of core concrete interaction with cavity walls on the Containment integrity.

e) High Pressure Core Melt Ejection

- A high pressure core melt ejection is the ejection of core debris and hydrogen from the reactor vessel at high pressure. High pressure core melt ejection could cause fragmentation and dispersal of core debris and hydrogen within the Containment atmosphere, termed direct Containment heating (DCH) that has the potential to cause early Containment failure. Containment failure could occur due to the heat-up and pressurisation of the Containment as a result of hydrogen combustion and core debris heat generation. Another potential consequence of high pressure melt conditions could be a thermally induced failure of steam generator tubes while the RCS is at high pressure, leading to Containment bypass. The likelihood of failure of the steam generator tubes depends on several factors including the thermal hydraulic conditions at various locations in the primary system which determines the temperature and pressure to which the steam generator tubes are subjected as the Accident progresses. The presence of defects in the steam generator tubes will increase the likelihood of failure.

- The Design should include an AC-independent RCS depressurisation system for reducing the probability of high pressure melt conditions and the reactor cavity design should include features to enhance core debris retention in the reactor cavity (e.g., no direct pathway to the Containment atmosphere).

f) Containment Performance under Severe Accident Conditions

The Containment should be designed to have a high probability of withstanding the loads associated with Severe Accident phenomena. This should be done by demonstrating that the Containment will maintain its role as a reliable, low leakage barrier for approximately 24 hours following the onset of core melt accident. After 24 hours, releases from the containment should be controlled or ensure that a containment failure probability of 0.1 is achieved.

The Containment should be assumed to have failed if any of the following conditions occur:

- Containment structural failure
- The Containment is bypassed
- The Containment fails to isolate
- The Containment seal materials fail as a result of over-temperature or pressure
- The molten core debris melts through the concrete basemat into the subsoil

g) Severe Accident Management

The Design should include provisions to facilitate the management of Severe Accidents. This should include provisions such as instrumentation that can provide the operating staff with information on the Accident progression (e.g., parameter trends), provisions to supply water and electrical power from outside sources (e.g., fire protection system water, portable generators) and provisions to protect the operating staff from radiation and toxic gases such that they can safely perform the actions called for in the Accident Management programme. The Design provisions should be consistent with and support the NPP's Accident management programme.

h) Release of Radioactive Material

The annual risk to members of the public from the release of Radioactive Material from a Severe Accident should not exceed the risk equivalent to a Dose of 1 mSv/yr. Appendix A provides guidance on the methodology to be used in calculating the annual Effective Dose to members of the public.

Documentation

Article (13)

The Licence application should address how the evaluation criteria contained in Articles (5) through (12) of this guide are met.

References

Article (14)

1. FANR-REG-03, “Regulation for the Design of Nuclear Power Plants”
2. FANR-REG-05, “Regulation for the Application of PRA at Nuclear Facilities”
3. FANR-REG-04, “Regulation for Radiation Dose Limits and Optimisation of Radiation Protection for Nuclear Facilities.”
4. NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants,” Revision 3, U.S. Nuclear Regulatory Commission, March 2007.
5. USNRC Regulatory Guide 1.196, “Control Room Habitability at Light-Water Nuclear Power Reactors,” Revision 1 (January 2007).
6. NUREG-1465, “Accident Source Terms for Light-Water Nuclear Power Plants,” (U.S. Nuclear Regulatory Commission – February 1995).
7. Title 10 of the *Code of Federal Regulations* (10 CFR) 50.61, “Fracture Toughness Requirements for the Protection Against Pressurised Thermal Shock Events,” U.S. Nuclear Regulatory Commission.
8. NUREG-1524, “A Reassessment of the Potential for an Alpha-Mode Containment Failure and a Review of Broader Fuel-Coolant Interaction Issues,” U.S. Nuclear Regulatory Commission, August 1996.

Appendix A

Calculation of Annual Risk to Members of the Public from the Release of Radioactive Material from Severe Accidents

1. The annual risk to members of the public is expressed as an annual Effective Dose to the Representative Person off-site.

2. The Representative Person off-site should be assumed to be an unsheltered individual located at the site boundary for a release duration of two hours during the Accident. If credit is taken for evacuation, the duration should be consistent with Emergency Preparedness assumptions.

3. The annual Effective Dose calculation should be based upon results from the NPP Design and site specific PRA with respect to the frequency, magnitude and timing of releases of Radioactive Material to the atmosphere due to Accidents and the site specific meteorology. All modes of plant Operation should be considered. Credit can be taken for decontamination of the Radioactive Material prior to release provided that sufficient justification is provided. Mean values for each of the parameters (e.g., release frequency, release fractions, meteorology) should be used.

4. If more than one NPP is located on the site, then the contribution to the Dose from multiple Facilities should be assessed for those events that can simultaneously affect multiple Facilities.

5. In accounting for wind direction, each of the wind directions categorised in the site-specific meteorological data (called sectors) needs to be assigned a probability based on the data. This probability should then be used in the annual Effective Dose calculation.

6. The annual Effective Dose to the most Representative Person in each sector should then be calculated using the frequency weighted Radioactive Material releases from the PRA, the average wind speed in each sector and the probability of the wind direction being in that sector. Within each sector there will be a Dose distribution across the sector and the peak Dose from that distribution should be used in the calculation of annual Effective Dose.

7. The annual Effective Dose to the most Representative Person in each sector should then be compared to the target Dose criterion of 1 mSv/yr.