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Seismic design considerations of nuclear fuel cycle facilities

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FOREWORD

In recent years, large earthquakes have affected nuclear fuel cycle facilities, such as Hyogoken-Nanbu (Kobe) in 1995, raising the issue of the seismic design of nuclear fuel cycle facilities and particularly of its consistency with the safety criteria developed for nuclear power plants.

Typically, fuel cycle facilities have been constructed over the last 40 years incorporating various seismic design provisions. These provisions were formulated at different times and range from no specific seismic requirements to those based on national building codes (aimed at structural integrity in case of an earthquake) to the latest seismic nuclear codes (aimed at compliance with radiological safety criteria in case of an earthquake).

An Advisory Group Meeting (AGM) on Seismic Technologies of Nuclear Fuel Cycle Facilities was convened in Vienna from 12 to 14 November 1997. The main objective of the meeting was the investigation of the present status of seismic technologies in nuclear fuel cycle facilities in Member States as a starting point for understanding of the most important directions and trends of national initiatives, including research and development, in the area of seismic safety.

Recommendations were also requested from Member States to assist the IAEA in the formulation of future IAEA assistance programmes.

The AGM gave priority to the establishment of a consistent programme for seismic assessment of nuclear fuel cycle facilities worldwide.

A consultants meeting (CS) subsequently met in Vienna from 16 to 19 March 1999. At this meeting the necessity of a dedicated programme was further supported and a technical background to the initiative was provided.

This publication provides recommendations both for the seismic design of new plants and for the re-evaluation projects of nuclear fuel cycle facilities. After a short introduction on the general IAEA approach, some key contributions from Member State participants are presented.

This publication complements the safety concepts developed for nuclear power reactors (see for example IAEA Safety Standards Series No. NS-R-1, *Safety of Nuclear Power Plants: Design*) while it recognises the reduced, but still significant hazard associated with fuel cycle facilities in line with companion publications such as IAEA-TECDOC-348, *Earthquake Resistant Design of Nuclear Facilities with Limited Radioactive Inventory,* dedicated to seismic design.

The IAEA wishes to express its gratitude to the AGM and CS chairman, J.K. Asmis and all the participants. The IAEA officers responsible for the organization of the meetings and for this publication were Y. Orita, N. Ojima and K. Kawabata of the Division of Nuclear Fuel Cycle and Waste Technology, and P. Contri of the Division of Nuclear Installation Safety.

EDITORIAL NOTE

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1. INTRODUCTION

1.1. Background

1.1.1. General

The term nuclear fuel cycle facilities (FCF) essentially refers to those facilities (other than nuclear reactors) which are meant for fuel fabrication, spent fuel storage, fuel reprocessing and waste management. The Advisory Group meeting (AGM) of November 1997 identified a wide range of plants, premises and installations that could be considered to be part of the nuclear fuel cycle. These include:

- Mines
- Fuel conversion plants
- Enrichment plants
- Fuel fabrication plants
  - natural U-fabrication plants
  - enriched U-fabrication plants
  - MOX fabrication plants
- Storage of irradiated fuel
  - dry storage
  - wet storage
- Reprocessing plants
  - decanning
  - separation
- Waste storage and handling facilities
  - for gaseous waste
  - for liquid waste
  - for solid waste :
    - silos
    - cans
  - compaction plants
  - vitrification plants
  - incinerators
- Other plants (sometimes included into FCF)
  - hot cells
  - accelerators
  - research facilities related to the nuclear fuel cycle
  - heavy water plants.

The majority of existing nuclear fuel cycle plants are in operation, but a significant number of new facilities is under construction particularly in the area of fuel storage and fuel reprocessing.

It was also observed that the existing plants may be subjected to modification, extension or adaptation.

1.1.2. Definition of front-end and back-end facilities

The main activities in the nuclear fuel cycle facilities comprise the following: mining, milling, refining and conversion, enrichment, fuel fabrication, spent fuel storage and reprocessing, waste storage, treatment and disposal and associated research facilities. The scope of this report does not include nuclear power plants.

There are two possible configurations of the fuel cycle:

a) The “once-through” fuel cycle, in which the ore is used to make fuel which is passed through the reactor once and is then stored in a facility waiting for the final disposal.

b) The reprocessing cycle, in which the fuel is passed through the reactor, stored and reprocessed and passed through the reactor again.

The difference between the “once through” cycle and the reprocessing cycle is that the reprocessing cycle makes more efficient use of the fuel through extraction of Pu (Plutonium) and recycling of U^{235} (Uranium-235).

In this publication, the facilities related to mining, milling, refining and conversion, enrichment and fuel fabrication are regarded as the front-end facilities.

The back-end facilities are related to the spent fuel storage, spent fuel reprocessing, waste storage, treatment and disposal, etc.

1.1.3. Unique features of the nuclear fuel cycle facilities

There are many features which are unique to these facilities, particularly in comparison with nuclear power plants, such as:

(i) the lower pressure and temperature operations,

(ii) the criticality hazard during all the normal and abnormal events,

(iii) the irregular shape of civil structures, which may require an in-depth study of torsional effects,

(iv) the high degree of confinement requested during all the operational states and during abnormal events such as an earthquake,

(v) the provision of multiple containment or confinement barriers to control the spread of radioactivity,

(vi) the shielding considerations giving rise to heavy structures,

(vii) the associated corrosion, erosion and ageing problems under toxic and radiation environments,

(viii) the planning for long-term repair times,

(ix) the associated chemical and fire hazards during their operation (industrial hazard)

(x) the special care during component layout considering the repair, maintenance and replacement strategies in the future,

(xi) the low energies associated with the radioactive inventory,

(xii) the importance of long-term equipment operability and structural integrity, the importance of maintaining long-term passive but secure containment or confinement (e.g., storage tanks, reservoirs),

(xiii) the potential for nuclear waste, spent fuel and material handling accidents

(xv) importance of ventilation systems as part of the confinement barriers.

These peculiar features are at the base of the specific safety considerations and the design recommendations in the next chapters.

1.1.4. Other IAEA programmes in related fields

In the field of nuclear safety, design of FCF structures is not covered yet by a dedicated series of IAEA Requirements document or Safety Guides. The basic reference is the Safety Fundamentals. However a series of technical documents is available covering specific aspects, such as seismic design (IAEA-TECDOC-348), accident reporting (INES Data Base), siting (IAEA-TECDOC-403–416).

A general initiative of the IAEA Department of Nuclear Safety has been in progress since 1999, based upon a comprehensive collection of the state-of-the-art regulations in Member States. Its aim is the development of a specific Requirement document for FCF, in addition to the Requirements for NPPs and Research Reactors, and a related series of Safety Guides dealing with:

- Mining, milling and refining
- Conversion and enrichment
- Fuel fabrication — uranium
- Fuel fabrication — MOX
- Reprocessing

This TECDOC represents an authoritative contribution to this development process and, more in detail, to the revision process of IAEA-TECDOC-348, in progress at NS, which will provide detailed procedures for the seismic design of FCF.

1.2. Objectives

The AGM and the CS collected a state of the art in the experience of Member States in the seismic design of these facilities as a starting point for a discussion on common safety objectives and therefore seismic design principles.

The approaches presented at the AGM as experience in the Member States have some differences among them, but they represent a necessary technical basis for any further steps, both in the derivation of seismic design principles and also in general safety objectives with reference to generic external events.

The main target of this publication is the presentation of these approaches in view of the completion of the general programme requested to support the design of the fuel cycle facilities. In some cases a synthesis has been tried among them as a fall out of the technical discussion and the results are described in the introductory part of this publication.

1.3. Scope

The discussion at the AGM concluded that mining and deep permanent disposal facilities of spent fuel and radio active wastes are beyond the scope of this review for their specific structural problems that could not be treated together with most of the fuel cycle facilities: therefore they deserve a dedicated discussion elsewhere. Also the nuclear power plants are discussed here only as reference plants where the design principles are in general more evolved and substantiated by application experience.

Moreover it was recognised that the fuel cycle facilities have very low standardisation design among Member States which creates some difficulties to any attempt of a synthesis. Therefore this publication tries to identify common features among such different plants with a final target in the seismic design, disregarding other big differences that mainly affect construction or operation.

1.4. Structure

The publication presents a short introduction on the IAEA approach and on the main outcome of the meeting, with an attempt of a unified approach to the seismic design of the fuel cycle facilities. Many representative contributions from the AGM participants are collected in the appendix in their original form.

2. NUCLEAR SAFETY CONSIDERATIONS

2.1. Design and nuclear safety

Similar to nuclear power plants (see IAEA NS-R-1 *Requirements for NPP Design*), the fuel cycle plants should be designed according to the “defense-in-depth” approach against the potentially significant failures, which could result in a release of radioactive materials to the environment. This can be achieved by:

a) Multiple physical and procedural barriers;

b) The provision of several levels of protection which prevent the breach of any barrier or mitigate the consequences of the breach;

c) Robust design and construction of containment or confinement systems.

It was recognised that Member States have different safety objectives relating to nuclear facilities for the nuclear fuel cycle, very often different from the criteria applied for nuclear power plants. However, some basic aims are common, such as the containment and confinement of radioactive material[1]and the protection of the work force and the public from the effects of ionizing radiation and toxic substances, as identified for the nuclear plants in general.

Each country establishes its own guidance for the seismic hazard, earthquake design and mitigation measures which lead to an overall assessed risk. Targets can be found by comparison to international experience and from best engineering practice. The quantity, form and isotopes in the plant's inventory, i.e. the magnitude of the radiological hazard, should be considered in the seismic hazard evaluation.

In many countries, the evaluation of such radiological hazard related to the fuel cycle facilities leads to the so-called "graded approach"[2] to the design. An example of application is provided in Table I, according to the experience of one Member State. The main idea of such approach is that a more simplified (i.e. higher conservatism) seismic methodology could be used for lower risk facilities, either with low inventory or with lower unmitigated hazard consequences. Table I presents an example of radiological hazard criteria which could be used to categorize hazardous facilities and systems, structures and components into various safety and design classes. The evaluation of the seismic capacity of the safety related systems, structures and components can be carried out consistently according to specified procedures with different levels of conservatism.

The safety class defines the relative importance of structures systems and components to nuclear safety. They are typically identified as safety class 1, safety class 2 and NNS class. The design class as a function of Facility Hazard Category and safety class establishes the basic seismic demand requirements. Within each design class is the potential for one or more Acceptance Criteria (I, II, III) to be used.

The acceptance criteria are contained in any design category which establishes the structural behaviour state (e.g. elastic, inelastic, etc.) that would be permitted for structures, systems and components. Acceptance criteria are established for example in IAEA TECDOC 348 on Earthquake Resistant Design of Nuclear Facilities with Limited Radioactive Inventory as Category I, II and III.

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[1] Containment is distinguished from confinement in that it implies a pressure retaining function (i.e. greater than 35 kPa) while confinement does not.

[2] For a discussion of graded approach related to seismic and other external natural events reference can be made to US Department of Energy documents: DOE-STD-1020-94 (Natural Phenomena Hazards Design and Evaluation Criteria for Department of Energy Facilities), DOE-STD-1021-93 (Natural Phenomena Hazards Performance Categorization Guidelines for Structures, Systems and Components), DOE-STD-1022-94 (Natural Phenomena Hazards Characterization Criteria), DOE-1023-95 (Natural Phenomena Hazards Assessment Criteria), DOE-1024-92(Guidelines for Use of Probabilities Seismic Hazard Curves at Department of Enregy Sites for Department of Energy Facilitiers), DOE-1027-92 (Hazard Categorization and Accident Analysis Techniques for Compliance with DOE Order 5480.23, Nuclear Safety Analysis Reports)

TABLE I. RELATIONSHIP BETWEEN HAZARD CATEGORY OF FACILITIES AND SAFETY DESIGN CLASSES OF STRUCTURES, SYSTEMS AND COMPONENTS WITHIN THOSE FACILITIES

DESIGN CLASS SELECTION

Hazard Category / Safety class	High 1	Moderate 2	Low 3	Conventional 4
Safety class 1	Design class 1	Design class 2	Design class 3	Design class 4
Safety class 2	Design class 2	Design class 3	Design class 3	Design class 4
NNS class	Design class 4	Design class 4	Design class 4	Design class 4

The performance goal is established by national regulatory authorities usually in the form of probability of undefined release of radioactivity to the public.

The nuclear fuel cycle plants (e.g. waste storage) often have extended lifetimes. According to this graded approach, this fact should be taken into account in the safety considerations, e.g. maintenance regimes, design return periods and degradation, and an opportunity should be given, where possible, for plant improvements to be carried out during the lifetime of the facility.

Essential services should be provided to ensure that a safe plant state both during and after a seismic event is maintained. Where essential plant services are shared with other plants on a multi-plant site, the effect of the sharing should be taken into account in assessing the adequacy.

Potential hazards developing from a seismic event should be identified, e.g. fire. The design concept should be such that the sensitivity of the plant to a seismic event is minimized. Nuclear fuel cycle plants normally have low temperature and pressure conditions. Exothermic or high pressure reactions should be avoided, and any source of energy released into the system should be adequately controlled as should the state of the nuclear matter in the plants. The use or generation of hazardous or toxic materials should be avoided where possible. The materials used for plant system components should be capable of withstanding the effects of corrosion and erosion.

Emergency plans should be capable of extension and dealing with a number of simultaneous plant failures as well as chemo-toxic, fire and radiation hazards.

2.2. Seismic hazard considerations

In addition to the efforts spent in the improvement of engineering procedures, there have been parallel efforts in the understanding and better defining the seismic hazard. The seismic hazard is defined as the ability of the region surrounding the nuclear facility site to generate earthquakes that can consequently damage the plant. Damage to the plant may be caused by site (vibratory) ground motion, by surface displacements (e.g. fault, site slumping, slope failure) within the site and by consequential damage caused by, for example upstream dams, landslides, seiches and tsunami (see for example IAEA 50-SG-S *Requirements for NPP siting*).

It is the purpose of the seismic hazard analysis to estimate future earthquake activities and their damage potential at the site. A seismic hazard analysis will use all evidence available from the geo-sciences, from historical earthquake records to define the seismic potential. The hazard investigation may also include site specific studies to better understand the site and characterise the local and regional geology and their potential to contribute to the seismic hazard. It may also be desirable to activate a local seismic network to record, document and analyse low level earthquake activity as a confirmation of the other available data.

In general, it was recognised that a siting procedure for a fuel cycle facility has to follow the same approach suggested for the nuclear power plants. In case simplifications are inserted, their conservatism should be demonstrated.

Only in case the risk the facility poses to the worker, the public and the environment is considered similar to that defined in National Building Codes, the requirements of such codes may be used to define the seismic hazard. In all other cases, appropriate safety margins should be applied to the seismic hazard definition consistently with the potentially induced hazard by the facility.

3. SEISMIC DESIGN OF NUCLEAR FUEL CYCLE FACILITIES

3.1. Design

The design features relevant to nuclear fuel cycle facilities are mostly governed by the various processes contained therein. During the design of these facilities, a great emphasis is laid on the simultaneous examination of various considerations such as planning for the layout of equipment, the repair and maintenance strategies, the facility construction, operation and final decommissioning and decontamination. This is on account of the fact that once these facilities start functioning, it is often difficult to provide repair/maintenance work on these facilities because of the associated radiation fields. The dimensions of the civil structures in these facilities may be decided mostly based on radiation shielding considerations. These structures serve as protective barriers to mitigate against the external events such as an earthquake.

3.1.1. Categorization

The facilities in a nuclear fuel cycle encompass the fuel mining facilities, fuel fabrication facilities, spent fuel storage facilities, fuel reprocessing facilities, waste management facilities, waste storage facilities, etc. A rational way is to categorize these facilities based on the intended design objective of the facility (i.e. the performance goal) and the consequent risk associated with it in the event of a failure on structures, systems and components relevant to the facility. Based on these criteria, these facilities can be placed in four categories viz. general use facilities, low hazard facilities, moderate hazard facilities and the high hazard facilities.

Categories of facility hazards, viz. 1, 2, 3 and 4 as shown in Table I are a function of both the radiological inventory and the potential for the release of radioactivity. Specific quantities used for facility hazard categorizations are established by national regulatory agencies of Member States. For example[3], the conventional buildings in these facilities may be put under the general use facilities whereas the mission dependent essential facilities such as fire station, computer facilities, natural U fabrication facilities, etc. shall be put under the low hazard facilities. The moderate hazard facilities, such as the uranium enrichment plants, spent fuel storage, etc., require the confinement of contents for the protection of plant personnel as well as the public. The high hazard facilities should be designed as to have a high degree of confidence that the hazardous materials are confined both during and after the occurrence of a natural phenomenon such as a seismic event. Although a facility may not contain hazardous materials, its function may be required during or after a seismic event, e.g. fire station, emergency power generation, pump houses. To provide the emergency services needed the fire station can be given an equivalent hazard category 3 and design class 3. One of the examples of hazard categorization of facilities may be found in Table I.

3.1.2. Design levels and acceptance criteria

The structures, systems and components in such facilities are classified for their seismic design according to their function and the degree of integrity required for plant safety. The definition of required seismic levels for their design depends on the performance goal and the annual hazard exceedance probability for the Structures, Systems and Components (SSC) within the facilities. For example, the performance goal exceedance probabilities are of the order of general use $1x10^{-3}$, low hazard $1x10^{-4}$, medium hazard $1x10^{-5}$, and high hazard $1x10^{-6}$ respectively. The associated annual seismic hazard exceedance probabilities for the SSC can be of the order of $2x10^{-3}$, $1x10^{-3}$, $5x10^{-4}$, and $1x10^{-4}$ respectively. Seismic hazard probabilities and performance goals are generally established by the Member States. The differences between these two probability levels indicate that enough conservatism shall be introduced in the seismic design or evaluation of these facilities. These seismic hazard exceedance probabilities need to be computed for each facility separately as a function of its hazard category. Based on these probability levels, the seismic hazard for each facility needs to be defined which is consistent with the geology and seismology at the site.

3.1.3. Acceptance criteria I, II and III[4]

Within any design classification as shown in Table I there are a number of potentially different acceptance criteria. They range from essentially elastic response all the way to near collapse, instability or failure. The three suggested acceptance criteria and the design parameters of ductility or inelastic (i.e. energy) demand ratios F[5] or ductility coefficient, and associated damping values are discussed as categories I, II, III in Section 3.1.2, paragraph 305 of IAEA-TECDOC-348 on Earthquake Resistant Design of Nuclear Facilities with Limited Radioactive Inventory.

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[3] These categorizations are established by individual Member States.

[4] As indicated in IAEA-TECDOC-348 on Earthquake Resistant Design of Nuclear Facilities with Limited Radioactive Inventory

3.1.4. Design codes/standards

The four facility hazard categories identified in Table I. typically contain 3 safety class SSC and are further classified into 4 design class SSC. The 3 safety classes are defined as safety class 1, safety class 2 and Non-Nuclear-Safety (NNS) class. The safety class 1 SSC are those which are designed to prevent, mitigate, contain or confine the consequences of a design basis event. Safety class 2 SSC are those which provide services such as lubrication, cooling, power, instrumentation and control[6] to safety class 1 SSC. Class 2 may also be used when the unmitigated event does not exceed national guidelines relative to the public or environment outside the facility area, but may have some relevance to operators and workers within the facility.

Based on the hazard categorization of a facility, and the safety classification of the SSC in the facility, there are 4 design classes for these SSC. A suggested relationship between Hazard Category of facilities and safety class and design class of SSC within these facilities is shown in Table 1. Design class 1 would be the highest class SSC available. Associated with each design class are quality processes and procedures. An example correlation of design classes is the quality class breakdown used in the U.S. from Regulatory Guide 1.29 where design class 1 would correspond to quality class A or B, design class 2 with quality class B, design class 3 with quality class C and design class 4 with quality class D as shown in Table II.

The choice of design codes/standards for the design of structures, systems and components pertaining to these facilities depends on their intended function and on the resulting consequences in terms of the radiological risk induced in the event of their failure. The choice of design codes/standards, therefore, has to be consistent with their design classification.

Quality class: In the construction of nuclear facilities a quality class designation is often used to define the applicable code or standard to be used.

3.1.5. Structural design

3.1.5.1. Methods

The seismic design of design class 1 and 2 structures is performed following conventional methodologies used in the nuclear power plant industry.

For the analysis, the following methods are used:

- static — equivalent
- dynamic-modal spectral
- dynamic time history

with more or less sophistication. In practical applications, a beam-type (simplified) dynamic model, if necessary, is usually followed by a detailed 3 D static finite element model, or alternatively a complete 3 D finite element dynamic model. Models have to take into account the various features of nuclear facilities structures such as plant irregularities, incidental torsion, etc. In the latter case, some additional eccentricities may be introduced, following national seismic design codes provisions.

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[5] For further explanaiton see IAEA-TECDOC-348 on Earthquake Resistant Design of Nuclear Facilities with Limited Radioactive Inventory.

[6] It should be noted that emergency power, instrumentation and control support to safety class 1 SSC are also often identified as safety class 1.

TABLE II. ILLUSTRATIVE QUALITY AND DESIGN STANDARDS

Components	Quality A Design class 1	Quality B Design class 2	Quality C Design class 3	Quality D Design class 4
Pressure Vessels[(1)]	ASME B&PVC Section III "Nuclear Power Plant Com-ponents," ASME class 1	ASME Boiler and Pressure Vessel Code, Section III, "Nuclear Power Plant Com-ponents," ASME class 2	ASME Boiler and Pressure Vessel Code, Section III, "Nuclear Power Plant Components," ASME class 3	ASME Boiler and Pressure Vessel Code, Section VIII, Division 1
Piping[(1)]	As Above	As Above	As Above	ANSI/ASME B31.1 Power Piping
Pumps	As Above	As Above	As Above	Manufacturers Standards
Valves[(1)]	As Above	As Above	As Above	ANSI/ASME B31.1
Atmospheric Storage Tanks	------	As Above	As Above	API-650, AWWA D 100, or NSI B 96.1
0-1 Atm. Storage Tanks	------	As Above	As Above	API-620
Structures	ACI 349 AISC N690	ACI 349 AISC N690	ACI 318 AISC Bldgs.	ACI 318 AISC Bldgs.
Electrical Equipment	Class 1E IEEE 323 IEEE 344	Class 1E IEEE 323 IEEE 344	Commercial Stds.	Commercial Stds.

(1) ASME B&PVC Section VIII Division 2 or B31 requirements as applicable are sometimes substituted for Section III with suitably augmented administrative (documentation) and Quality Assurance requirements when procurement of Section III Components becomes impractical.

3.1.5.2. Siting

For design class 1, the seismic siting and design procedures contained in IAEA 50-SG-S1 *Requirements for NPP siting* and 50-SG-D-15 *NPP Seismic Design* respectively are recommended. For design class 2 the procedures contained in IAEA-TECDOC-348 on Earthquake Resistant Design of Nuclear Facilities with Limited Radioactive Inventory are recommended. For design classes 3 and 4 national building codes for conventional hazardous and ordinary facilities respectively are recommended.

3.1.5.3. Structural acceptance criteria

The structural acceptance criteria used for design classes 1 and 2 are generally those associated with the conventional limit or strength state with load factor taken equal to 1.0. However, particular emphasis should be given to safety function, leak tightness and containment where limitation of radiological release is the prime safety goal which often requires more stringent structural acceptance criteria than simply the avoidance of structural failure.

The structural acceptance criteria for design classes 3 and 4 are usually limited to that prescribed in national building codes.

3.1.5.4. Inelasticity

For design, an excursion into the inelastic domain is a function of acceptance criteria I, II and III. For design class 1, elastic demand is usually required as per acceptance criteria I. For design class 2, and acceptance criteria II and III, limited excursions associated with global ductility ratios between 1 and 3, in ductile structures, may be permitted. For existing design class 1 and all design class 2 evaluations limited excursions into the inelastic domain are allowed for ductile structures in accordance with national practice. New design classes 3 and 4 and acceptance criteria I, II and III generally follow current national seismic practice for essential or conventional hazardous or ordinary structures, systems and components.

For design classes 3 and 4 inelasticity coefficients associated with national building codes are usually permitted.

3.1.6. Mechanical and electrical equipment design

A wide range of facilities and equipment are employed in nuclear fuel cycle facilities. Some of the equipment have functions which are critical for establishing and maintaining a safe state during/after the occurrence of an earthquake. Typical equipment which are required to perform or have seismic safety related functions can be grouped as:

1) Containment equipment which is intended to prevent or limit the spread of radioactive or chemically hazardous materials;

2) Mechanical equipment which are employed to transfer hazardous materials between process locations, are involved in plant maintenance or recovery from abnormal events or have other safety related function;

3) Control and instrumentation equipment which are employed to identify the condition of the plant or process and maintain or return it to a safe state;

4) Storage systems which house or support containment items;

5) Services equipment which are necessary to maintain the process in a safe state or are required to facilitate recovery from the effects of an earthquake.

Typical equipment for the above categories are:

1) Tanks, vessels, piping, glove boxes, ventilation systems;

2) Cranes, mechanical handling equipment, shielding units, shield doors;

3) Temperature, pressure and flow rate sensors and associated relaying and indicator systems, shut-off valves (manual or automatically triggered by an earthquake);

4) Storage drum stacks, canning and racking systems;

5) Electrical systems for lighting and control systems, firefighting systems, accident or emergency recovery equipment, cooling water systems.

In all the cases, it is highly desirable that the functions required of the various mechanical equipment be established at an early stage in the overall development of a facility. It is usual for seismic design categories to be employed to aid the description and communication of the requirements. The most cost effective mix of design requirements may be determined by effective discussion between the specifiers of safety systems and the mechanical equipment and building structure designers.

The functions required of mechanical equipment may vary according to the size and frequency of earthquake. In some areas, potentially damaging earthquakes are considered to occur with a frequency which requires a facility to be operable after an earthquake. Mechanical equipment design conditions to meet such requirements can differ from those employed to ensure that the facility is in a safe state following a large, less frequent event.

All the systems and equipment are classified into four design classes according to the definitions above. Requirements imposed on equipment are integrity, operability or stability. Acceptance criteria should be defined for each case.

The wide variety of mechanical equipment which has a seismic design requirement leads to a wide variety of criteria being used to determine their capacity to resist earthquake loading. The acceptance criteria are often based on existing codes of practice from countries where provisions relating to earthquake loading are included in the codes, or on codes familiar to the designers of each class of equipment.

Alternatively, acceptance criteria may be based on shake table testing or the use of appropriately valid preexisting data on the seismic performance of equipment. Shake table testing is particularly applicable to whole plant and equipment systems for which calculations for individual components of the equipment is not practicable or where design calculations and acceptance criteria may lead to overly conservative designs. Shake table testing is often carried out on electrical equipment (switchgear, motor controls, transformers, relay system, etc.).

The earthquake loadings for design of equipment are typically determined with due consideration of the behaviour of the building structures in which the equipment is housed. For some large and heavy equipment, it may be preferable that the mechanical equipment is incorporated into the assessment of the behaviour of the building structures. Equipment seismic design loads are usually static equivalent accelerations, response spectra and time histories. In some cases, relative displacement loading may also need to be considered.

Equipment analysis techniques vary according to the description of the earthquake loading available to the designers, the parameters required to be determined for comparison with the acceptance criteria and the degree of precision required in the calculations to ensure a cost effective design consistent with the potential hazard.

3.2. Testing

In the process of seismic qualification of structures, components and equipment, testing is used in the following different ways. In this respect, the IAEA safety guides such as 50-SG-D-15 are useful as a reference.

1) For complete qualification of an equipment which is tested at scale 1, to real earthquake excitation in order to verify the requirement of its safety function.

2) For the qualification (or verification) of an analysis methodology. In this case, a part (typically the most critical) or a mock-up of the structure or component is tested in order to quantify the behaviour. The overall seismic qualification is then obtained by analysis.

3) For studies on margins or for PSA, fragilities are necessary. They can be obtained either directly by test, similar to qualification tests as given in 1) above, but with increasing input level until failure occurs, or by combined test and analysis, as given in 2) above.

The most convenient device is a shake table which permits the simulation of a support type movement. The following points must be considered during the testing:

- Type of excitation: 1D, 2D or 3D;
- Careful representation of the supporting structure and anchorage, for equipment;
- Environment of the equipment: pressure, temperature, piping loads, presence of electrical power, etc.

In-situ tests on actual structures or equipment can be useful to quantify some parameters, such as boundary conditions, mode shapes, frequencies, etc. The test shaking level is very low if the equipment is required for further use. This prevents parameters such as damping from being quantified by such tests.

In-situ high amplitude tests (by strong shakes or explosives or actual earthquakes) can be applied on a R&D basis.

Typical examples of testing performed for nuclear fuel cycle facilities are:

- Storage drums, in order to determine the best seismic storage geometry (Japan, France, UK);
- Structural elements such as shear walls (fragility test in Japan, France, U.S. and Russia);
- Glove boxes (scale 1 in Japan).

3.3. Maintaining the design basis

Provisions of designs which are considered to counter the safety threat posed by earthquakes is the first step in ensuring seismic protection. The second step is to verify that the facility as constructed has no features which could detract from the designed case. For building structures the minimum standard of construction/design employed should be similar to the original design. However, for plant and equipment (including services equipment), as adjacent items may be of different types or designed by different mechanical designers it is preferable that an examination of the as-constructed facility is carried out. This is to ensure that there are no potential interactions between equipment or equipment and building structure items, all necessary seismic design features have been included, and no features are introduced which could unacceptably reduce the seismic protection. It is expected that this second stage would comprise a 'walkdown' of the as-constructed facility by reasonably experienced seismic engineers who would maintain appropriate records of their findings. Timely remedial action to bring the facility to an acceptable level of seismic protection is expected to ensure that the facility is essentially in the as-designed condition.

The final stage of maintaining the seismic design basis occurs once a facility has commenced active operation. It is possible that some actions may be taken which might reduce the as-designed level of seismic protection impacting various features such as those covered in para. 1.4. above. These actions would typically be carried out in order to ease or speed up the operation of the facility. Periodic reviews by reasonably experienced seismic engineers working in conjunction with safety assessment and plant operators should be carried out to ensure that a sufficient level of seismic protection is maintained. Remedial action should be taken to return and maintain the facility to a sufficient level of seismic protection when deficiencies are identified.

3.4. Lessons learned from the past earthquakes

Our knowledge concerning seismic technologies has progressed in part based on the experience gained from the past earthquakes. Fortunately, a significant accident from a nuclear fuel cycle facility has not been reported in past earthquakes. But, minor hazards such as power disruption, release of gases to the environment, have been reported.

In recent years two major earthquakes have been experienced; the Northridge earthquake 1994 in California and the Kobe earthquake in 1995 in Japan.

Nuclear fuel cycle facilities have features which are identical to the ordinary chemical industrial facilities. Hence, many lessons may be learnt from the earthquakes mentioned above, for example:

- Occurrence and spreading of fires;
- The effect of shutdown of lifeline systems; (Electric power, gas supply, water supply, sewerage systems, information networks, communication systems)
- Human losses caused by spreading of fire and collapse of houses
- Failure of piping systems and tanks;
- Emergency counter-measures (Activities by the fire department, traffic control, information connection, communications).

Lessons may be learnt both from where failures have occurred and the systems which have successfully withstood the earthquakes.

4. RE-EVALUATION OF EXISTING NUCLEAR FUEL CYCLE FACILITIES

For the existing facilities not designed to current standards, seismic re-evaluation may be needed for the following reasons:

- Improvement in the knowledge of local seismicity;
- Improvements considered in modern standards of seismic design.

The different steps in the process of re-evaluation are as follows:

1) Determination of the "as-is" condition of the plant, in order to have the necessary information for re-analysis and to identify any problems that might have decreased the seismic resistance (such as aging, corrosion, erosion, anchorage, spatial interaction, etc.).

2) The seismic hazard reference level against which the plant is reviewed takes into account the inventory, the remaining operational life of the plant, modern performance standards, and the safety significance.

   Inelastic behaviour can be accepted provided it can be demonstrated that the inelastic deformation is limited and that minimum structural detailing (e.g. stirrups, ties and connections) is present.

   The analysis method may consider inelastic behaviour where appropriate; some refined analyses may be needed for the assessment. Seismic experience data can be used to qualify equipment.

3) Assessment of the facility and identification of retrofits, if necessary. Decisions made must take into account their feasibility, the safety improvement and the cost.

Although the intention is always to reduce the overall risk potential of a plant, it is acknowledged that sometimes short term risk must increase for long term reductions.

5. FUTURE ACTIVITIES IN MEMBER STATES

A number of groundbreaking initiatives are in progress which will aid Member States to assure adequate seismic safety to fuel cycle facilities. These are summarised in the following for the information: their progress can be monitored and periodically brought to the attention of the Member States.

5.1. Testing by 3-shake table and relevant research

Japan is planning to construct the 3-D Full-scale earthquake testing facility. This facility has a table size of 20m x 15m, 1200 tons of loading capacity, 200 cm/s of maximum velocity and ±1m of maximum displacement. This facility allows the testing of full-scale structures and/or equipment under real earthquake conditions. By using this facility, we will get more practical and realistic information on structures and equipment performance during earthquakes.

Russia has had in operation a 30 m by 14 m 6D explosive test table supported by air bearings at Viborg. The capacity of this table for seismic testing is: 400 tons, 1 m/sec velocity and ±1 m displacement.

In the United States a 45 m by 45 m earth platform surrounded by a deep trench for the placement of explosive charges is currently under construction at a Nevada test site for use in large scale seismic simulations. Test capacities have yet to be determined.

5.2. Accumulation of strong-motion data

For the analysis of seismic safety of the structures and the equipment, we need more strong-motion data. In Japan, the K-NET (Strong-Motion Earthquake Observation Network) was constructed in 1996. This net consists of 1000 observation sites. The whole data set from K-NET (soil information of the site, time-history data, maximum acceleration, etc.) is distributed worldwide through the Internet. These data are useful to understand the characteristics of strong-motion such as the attenuation, response spectra and so on.

5.3. Exchange of testing project information

The tests for the nuclear fuel cycle facilities were conducted in France, UK and Japan. These countries also have a future testing programme. The exchange of testing project information should be encouraged.

5.4. Information exchange of R&D results

Continuous exchange of information is also very important and should be encouraged, in particular regarding the following topics:

- Seismic probabilistic safety assessment;
- Performance based design;
- Evaluation method for aged facilities.
- Existing seismic experience data bases are now largely used worldwide. An extension of this data base to incorporate local practices is encouraged.

6. FUTURE RESEARCH AND DEVELOPMENT NEEDS

- Obtain further views on the seismic qualification and performance of nuclear fuel cycle facilities from other Member States;
- Obtain information on the assessment of existing facilities for their seismic load withstanding capability;
- Develop criteria for applying simplified seismic analysis, design and assessment for plants which have a low overall risk.
- Provide guidance for establishing seismic hazard levels (return periods) which take into account the safety significance of the facilities.
- Creation of a database listing examples of facilities and their seismic design or re-evaluation. The database should include the various plant types and solutions adopted by Member States. The examples should include older facilities not originally designed for earthquake loading and the retrofitting measures taken; new facilities with both large and small inventories.
- It is suggested that the long term goal should be to create a safety guide in this subject area.

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PAPERS PRESENTED AT THE ADVISORY GROUP MEETING

SEISMIC DESIGN AND ANALYSIS OF NUCLEAR FUEL CYCLE FACILITIES IN FRANCE

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Abstract

Methodology for seismic design of nuclear fuel facilities and power plants in France is described. After the description of regulatory and normative texts for seismic design, different elements are examined : definition of ground motion, analysis methods, new trends, reevaluation and specificity of Fuel Cycle Facilities. R/D developments are explicated in each part. Their final objective are to better quantify the margins of each step which, in relation with safety analysis,lead to balanced design, analysis and retrofit rules.

1. DEFINITION OF THE METHODOLOGY

The methodology for the seismic design of french nuclear facilities has been established gradually for power plants since the beginning of the extensive PWR program in 1972. Regulatory Documents on seismic design were issued after 1981 in the form of RFS documents (Règles Fondamentales de Sûreté or "Fundamental Safety Rules"), when this experience was considered sufficient on a given specific subject.

So far, four of these seismic RFS documents related to seismic design and analysis have been published, namely :

- RFS 1.2.c : Determination of ground motions,
- RFS 1.3.c : Determination of soil characteristics,
- RFS V.2.g : Seismic Analysis of Civil Engineering Structures,
- RFS 1.3.b : Seismic Instrumentation.

In addition, three other RFS documents include information relating to seismic requirements concerning : classification rules (RFS IV.1.a), design and qualification rules, either for mechanical equipment (RFS IV.2.a) or electrical equipment (RFS IV.2.b).

In spite of their title, RFS documents are guidelines rather than regulations, as their observance is not strictly required ; they merely define acceptable procedures, which whenever applied, are considered adequate for licensing procedures. As can be seen from the list, the four seismic RFS documents do not cover the entire field of seismic design for NPPs ; a rule dealing with seismic analysis and design of equipment and piping, and a rule on load combinations are currently not dealt with as RFS.

Other technical documents, known as RCC ("Règles de Conception et de Construction" or Design and Construction Rules) have also been issued by the applicant ; they give a detailed description of design rules for all operating and accidental loads, including seismic loads, for structures (RCC-G), mechanical equipment (RCC-M) and electrical equipment (RCC-E). RCC-MR is applicable to mechanical equipment in F.B.R. These RCC documents are submitted to Safety Authorities.

The application of the requirements is documented by the applicant in the FSAR (Final Safety Analysis Report).

The above defined requirements apply to PWR, Fast Breeder and with adaptations to all nuclear installations such as reprocessing or experimental plants. Considering the seismic hazard, fuel facilities called laboratories and factories refers to the R.F.S. I.1.c, which is similar to the PWR counterpart R.F.S. I.2.c.

For conventional structures, a new seismic design code has been issued recently as a National Standard, PS92. It is comparable to recent seismic code in other industrialized countries such as Italy or USA. A noticeable difference is found in the design of reinforced concrete shear walls. The quantity and detailing of reinforcement is far less stringent than proposed in other codes such as the European EC8.

2. RE-EVALUATION

At present, there is no regulatory requirement for periodical updating of the FSAR. Nevertheless, there is a regulatory requirement (Decree 11 Dec. 1963, modified 19 Jan. 1990) for a general Safety Re-assessment. In this context, the advisory group of experts "Groupe Permanent" has recommended a seismic re-revaluation.

Several examples of seismic re-evaluation are known in France. They belong to two different types :

- seismic safety evaluation of existing facilities which have not been designed to withstand earthquakes, or which have been designed using old-fashioned rules ; this evaluation is aimed at defining the critical safety features and possible upgrading requirements.

- seismic re-evaluation of the seismic hazard at the site and its consequences for the safety of the facility.

The first type corresponds to some gas-graphite (CHINON-A2-A3, BUGEY) and fast-breeder (PHENIX) plant and some of fuel cycle facilities as in CADARACHE and LA HAGUE..

There is no regulatory requirements for periodic seismic re-assessment or re-evaluation. However, RFS 1.2.c stipulates :

Once irreversible decisions have been taken, where new data result in a stronger SMS spectrum, post-verifications may be performed by means of appropriate calculating methods for estimating safety margins against earthquakes. In this case, the estimation must be submitted to the administration for approval.

At present, there are only individual requirements for the seismic hazard re-assessment of several sites. These requirements were debated with the Safety Authorities.
EDF, COGEMA and CEA have started studies on seismic hazard re-assessment for their sites.

The rules that would be used for the re-analysis of the structures and equipment have not been presently defined. Usual rules are design rules and as such are not aimed at representing actual behavior but rather at leading to a design with significant margins. They should be adapted to seismic re-evaluation. In this way, capacity of simulating the non-linear behavior of structures is developed.

3. CURRENT FRENCH PRACTICE FOR SEISMIC DESIGN AND ANALYSIS

3.1. Determination of ground motions

This subject is dealt with by RFS 1.2.c for Power Plants or RFS I.1.c for fuel cycle facilities. The recommended method is deterministic and rely on the seismotectonic approach; this method is well suited to the seismicity situation in France where :

- historical seismicity is well documented,
- the seismicity level lies, in most regions, in the range " low to moderate",
- the intraplate character of the seismicity, in connection with its low intensity level, makes it difficult to identify, in most cases, the relation between observed earthquakes and know faults.

The ground motions for a given site are obtained by a several-step procedure :

1) Review of historical seismicity around the site.

2) Definition of tectonic domains from geological and seismic data.

3) Considering the occurrence of earthquakes similar to the historical earthquakes at the points of the tectonic domains which are closest to the site, definition of SMHV ("Seisme Maximal Historiquement Vraisemblable" or "Maximum Historically Credible Earthquake") which produces the highest intensity at the site.

4) Definition of SMS (Seisme Majoré de Sécurité), by conventionally adding one degree to the intensity of SMHV.

5) Computation of the free-field response spectrum for SMS using a magnitude, focal distance and intensity correlation and referring to a data-base. Its focal distance is smaller than 10 km (close focus earthquakes), a conventional spectral shape is used with an intensity acceleration correlation. This latter case can have a strong impact on design.

R/D is on going on this matter in way to, first, improve the knowledge of seismicity in France and second by accumulating strong-motion data to improve the data bank and by developing numerical simulation techniques.

A revision of the RFS 1.2.c has been proposed and accepted experimentally.

Main charges are related to progress achieved in the following domains :.

- techniques for characterization of seismotectonic zones and active faults,
- improvement in historical and instrumental seismicity of France,
- characterization of near-fault seismic motions,
- site effects.

3.2. Seismic analysis

Rules for seismic analysis are conventional ones used for N.P.P. with some features due to particular aspects of the French Nuclear program.

3.2.1. Soil Structure Interaction

For PWR, as a consequence of project standardization, it has to be emphasized that precise soil data for all sites are usually not available when design studies have to be conducted. Furthermore, these studies must take into account the variability of soil conditions in a wide range from soft soil to hard rock.

The analysis is performed taking into account rather simple SSI methods, but considering a significant variation of soil dynamic Young's modules. The definition of impedances in the case of separate foundations must be developed.

Main issues of SSI are related to the necessity of the validation of analytical techniques, such as the work performed or the Hualien experiment. The domain of application of simple methods must be assessed based on these results.

Uplift analysis and criteria have to be precised, and R/D is necessary for determination of soil characteristics to be used in the analysis and for assessing the best way to obtain them on site.

R/D is under way to quantify the effect of non-coherency of the input signal and the effect of wave passage. It is based on gathered on site measures, such as in Lotung and Hualien.

3.2.2. Structural analysis

Usually dynamic analysis using beam-type models allows the determination of global seismic loads ; the detailed structural analysis is performed one 3D finite-element model. For design, linear analysis is the preferred method ; non-linear calculation can be performed in some special cases.

During the process of modeling and analysis, many assumptions are made. Global qualification (benchmarks) should be developed in order to qualify the overall process.

Questions on whether or not behavior coefficients should be used in risen by reevaluation tasks. Ductility coefficients associated with actual detailing rules and special analysis techniques must be associated with the consideration of the non-linear behavior. Global non-linear elements are developed and validations tests are defined. Development of techniques to generate floor response spectra on non-linear structures must be conducted.

3.2.3. Equipment

There are designed using floor response spectra ; for PWR, they are defined as envelopes of spectra calculated with a large variation of soil condition as for structures (§ 3.2.2), to cover a large set of sites. For other plants, they are site related and have to consider the variation of soil properties, as defined in § 3.2.1.

Equipment are generally designed by conventional modal spectral analysis.

Some equipment have a non-linear behavior due to the presence of gaps, to sliding or to plasticity. There are analyzed by time history step by step analysis.

R/D programs are under way for the qualification and improvement of the non-linear behavior modelisation for fuel elements, cores, spent fuel racks, piping... and to gain a better understanding of their ultimate behavior. Experiment on shaking table are conducted in parallel with analyses.

An intensive R/D program is devoted to piping behavior in order to define more realistic criteria including the effect of the presence of defects.

For equipment whose functional capability is essential and cannot be demonstrated by analysis (mostly electrical equipment and some valves) seismic qualification is obtained by the shaking table tests. Qualification procedures usually rely on bi-axial tests using synthetic time histories.

3.3. Complements

3.3.1 Seismic approach

As pointed out above, resistance to earthquake is essentially the result of a deterministic design approach : seismic effects are considered as load cases, in combination with other design-basis loads. Nevertheless, a more realistic and accurate approach of the impact of an earthquake on the installation is progressively being applied. This new approach considers the earthquake as an initiating event which entails a comprehensive investigation into its potential damage on the various parts of the plant, leading to investigations such as :

– identification of the design-basis operating conditions which could be initiated by an earthquake, analyzing them accordingly with appropriate rules, such as using only operational equipment in the concurrence of a design-basis earthquake and postulating loss of off-site power.
– definition of operating requirements of equipment and systems assuring a safety function in the occurrence of an earthquake, taking account of potential plant failures, notably damage to non-safety-related equipment and their interference with safety-related equipment.
– reassessment of margins on the most vulnerable equipment with respect to earthquakes exceeding the level of the design-basis earthquake.

So far, only damage to non-safety-related equipment and their interference (usually through their fall) with safety-related equipment have been considered.

3.3.2. Seismic margins

As mentioned above and more generally, in order to gain a complete understanding of plant behavior and design in case of earthquake, margins studies are now beeing considered and adaptation of US practices (SMA) are looked at. Definitive methodologies to be used are under discussion. They must be related to the reevaluation topic.

4. SPECIAL FEATURES OF FUEL CYCLE FACILITIES

From structural point of view and having in mind the seismic behaviour, fuel cycle facilities have the following features.

- The main process and the highest quantity of radioactive material are often located in reinforced concrete boxes, with thick walls (typically around 1 m) for radioprotection.

Reinforced even to the minimum percentage, these boxes are usually naturally seismically resistant. For old facilities, the reinforcement may be very weak and justification of seismic behavior may require some complementary work.

- The main building structures are usually reinforced concrete shear walls, which generally have a good resistance to earthquake loads. Older facilities may have reinforced concrete frames (beam - columns) with masonry infills, which seismic behavior must be studied in cases of reevaluation ; retrofit may induce R&D on strengthening techniques.
- Fuel elements are stored in large pools ; large seismic pressures may lead to heavy structures. A seismic pads may lead to a more economic design. Leaktightness is usually achieved with a stainless steel liner. Storage rack must be stable during earthquake ; their analysis require fluid structure interaction, sliding and non-linear impacts-qualification of computer codes are needed with shaking table tests. Building above pools are very often steel structures which can be very easily be designed against seismic loads.

In fuel cycle facilities, building may be very irregular, with torsion and the earthquake may govern the design, contrary to LWR buildings for instance.

Components and equipment in facilities are comparable to nuclear power plants ; except some very specific such as glove boxes.

From a safety point of view, some "confinement" may be required during and after the earthquake. This is provided by special condition on structures (limited deformations during earthquake), by requiring operability of ventilation or leaktightness of gloves boxes and RC process boxes.

General seismic design methods are similar to these developed for power reactors, with comparable structural criteria. All the facility is analyzed in the same way. Gradual methodologies and criteria could be set up, according to radioactive inventory and safety conditions.

5. CONLUSIONS

Methodology for seismic design of nuclear fuel facilities and power plants in France has been reviewed. Main evolution are related, first, to a better knowledge of seismicity and seismotectonic of France and secondly to an improvement of the simulation of the (non-linear) structural behavior. These results in a quantification of seismic margins, which, associated to safety analysis taking into account the radioactive inventory, may propose a more balanced design for new facilities and data for reevaluation and retrofit of existing plants.

SEISMIC TECHNOLOGY OF NUCLEAR FUEL CYCLE FACILITIES: A VIEW OF BNFL'S APPROACH AND METHODS

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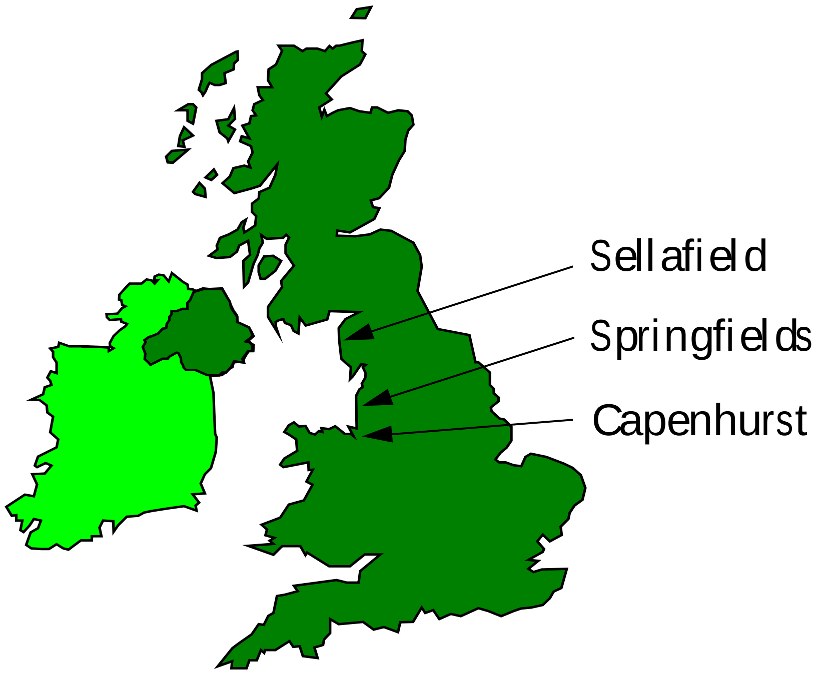
Abstract

The approach BNFL employs in the seismic qualification of its nuclear fuel cycle facilities is described in this paper. The overall seismic qualification process from design to installation and commissioning is considered. The approach for new facilities, such as the Sellafield Mixed Oxide Fuel Plant and Windscale Vitrification Plant Line 3 currently under construction, is examined.

1. INTRODUCTION

In addition to numerous reactor sites, BNFL own and operate three sites in the UK at which fuel cycle facilities are located (Figure 1). New facilities have been constructed recently at Springfields and Sellafield and a number of major new facilities are currently under construction at Sellafield. All three sites have numerous older facilities. The approach to seismic qualification of the newer facilities is reviewed in this paper. BNFL have interests in fuel cycle facilities in countries other than the UK through wholly or jointly owned subsidiary companies and through partnership arrangements with other companies. The approach to seismic qualification for these facilities is not considered here.



Notes:	Sellafield	- Reprocessing, storage and fuel manufacture.
	Springfields	- Fuel manufacture.
	Capenhurst	- Storage. Enrichment (URENCO).
	Reactor sites not shown.	

Figure 1. BNFL fuel cycle facilities locations in the UK.

The basis for the seismic qualification of a facility is composed of a number of components. These are:

1. The seismic hazard which describes the probability of earthquakes which might affect a facility.
2. The safety functional requirements of the various parts of a facility that intrinsically reduce or are explicitly provided to reduce the consequences to a tolerable level.
3. The design basis earthquake used in the design of the components of a facility.
4. The acceptance criteria by which it is judged whether the components of a facility will achieve their required earthquake safety function given the occurrence of the design basis earthquake.
5. The design methods used to determine the parameters that need to be compared against the acceptance criteria.

These five components are described in this paper.

The seismic hazard has not changed over a number of years so is seldom specifically revisited in detail in the development of the design of a new facility. Similarly the definition of the design basis earthquake has not changed over a similar period. The precise content of the other components is influenced by the form and function of facility and the procedures employed in design. The range of different processes employed are reviewed in the following sections.

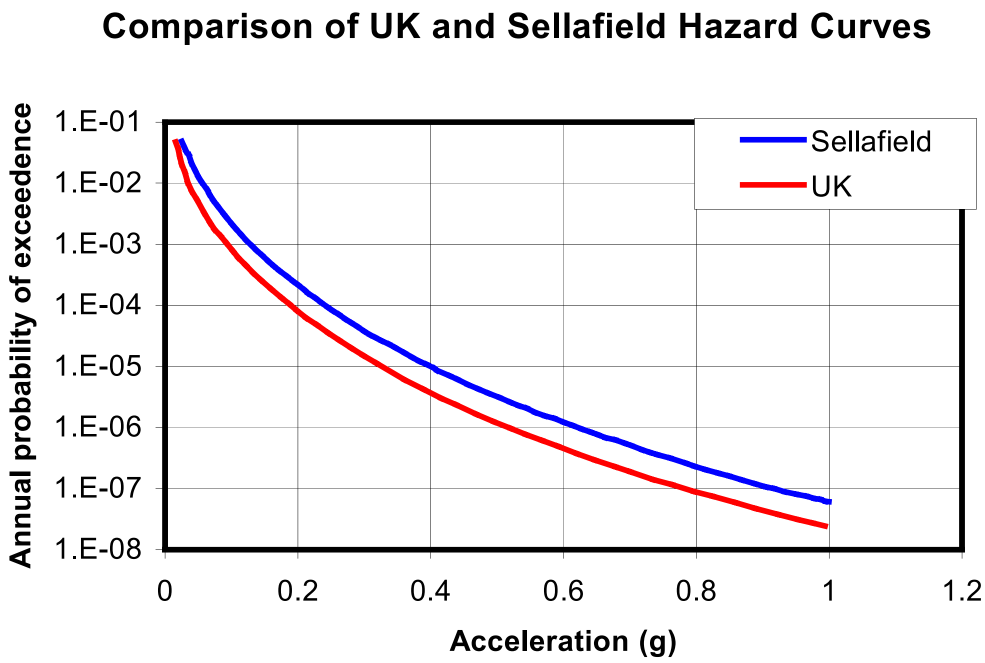
2. SEISMIC HAZARD

The seismic hazard for BNFL fuel cycle facilities is derived from work carried out in the early 1980s. This considered earthquake occurrence in the UK as a whole. Reports of earthquakes and of events which are now interpreted as due to earthquakes were examined from a wide range of historical records stretching back over a thousand years. Geological and tectonic information was also used in characterising the earthquake environment in terms of the peak free field horizontal acceleration (PFFHA) zero period acceleration (ZPA) and earthquake annual probability of occurrence. The characterisation obtained is known as the hazard curve.

Further examination of the area in the north west of England around Sellafield, the main reprocessing site now also used for fuel manufacture, developed the hazard curve for Sellafield. Figure 2 shows the hazard curves for Sellafield and the UK as a whole from these studies. The hazard curve confirmed that the 0.25g PFFHA ZPA already in use for design was conservative with respect to the 1 in 10,000 year earthquake. The 1 in 10,000 year earthquake tied to 0.25 g PFFHA ZPA remains the key design basis result from the seismic hazard studies.

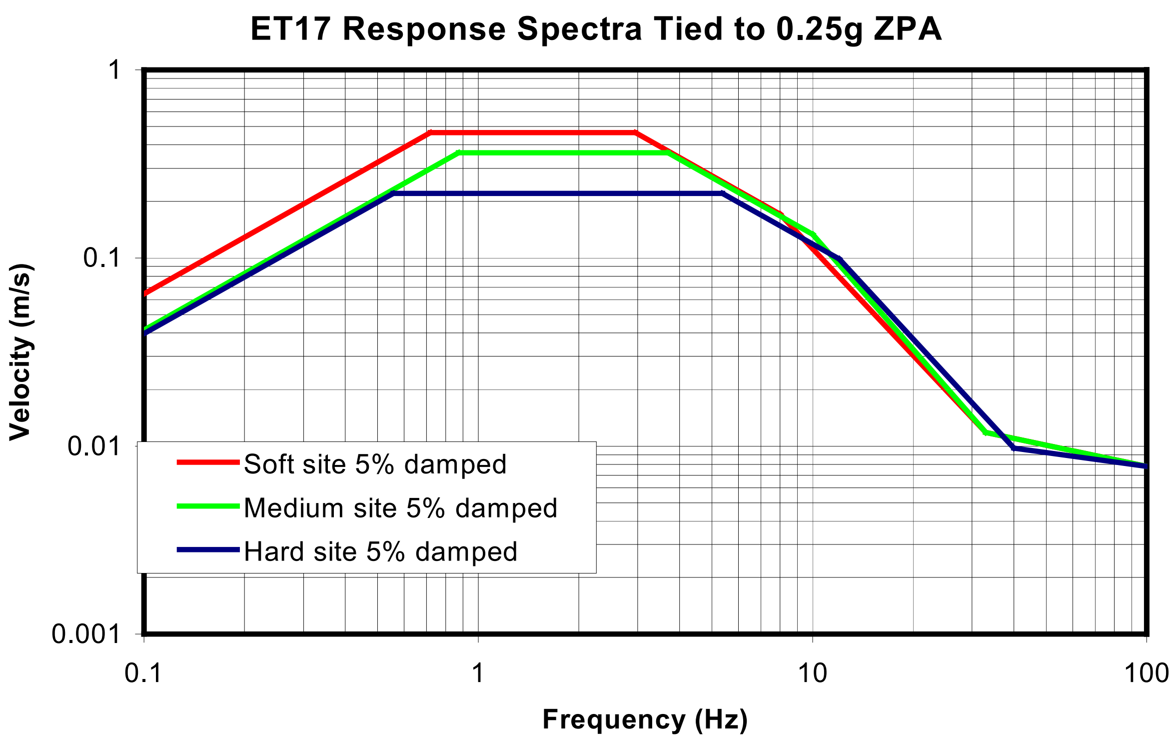
3. SEISMIC DESIGN BASIS

As well as the hazard curve the studies on the earthquake environment in the UK developed response spectra for hard, medium and soft soil sites. The spectra were derived from spectra of a large number of real earthquake records for seismic events of comparable size to that expected of the 1 in 10,000 year event. They included a conservative allowance for the statistical scatter of spectral values. The spectra developed are piecewise linear. They are defined with reference to displacement, velocity and acceleration limits (bounds) and corner frequencies. The bounded quantities vary with damping. Figure 3 shows the spectra for a typical damping value.



Notes: Acceleration is peak free field horizontal ground zero period acceleration.
1 in 10,000 corresponds to 0.24g ZPA rounded up to 0.25g for design.

Figure 2. Seismic hazard curve.



Notes: Spectra are tied to a zero period acceleration of 0.25g in figure.
The spectra may be tied to other values, e.g. 0.1, 0.125, 0.35g as required.

Figure 3. Ground free field horizontal response spectra.

The piecewise linear spectra were adopted for design of new facilities and assessment of existing facilities. In the horizontal direction for design of new facilities the spectra are tied to 0.25g ZPA and to 2/3 of this vertically. This definition gives the design basis earthquake (DBE). The spectra are also tied to 0.35g ZPA (horizontal) in reviews to ensure that new facilities would perform in a substantially similar manner to their design basis in earthquakes larger than the design basis event. When used for assessment of the seismic performance of existing facilities the spectra may be tied to other values of ZPA, typically a minimum of 0.125g in the horizontal direction and again to 2/3 of this vertically.

4. SAFETY FUNCTIONAL REQUIREMENTS FOR SEISMIC HAZARD

The safety functional requirements for a new facility are developed during the evolution of its functional and design specification to bring the effects of hazards, including the seismic hazard, to a tolerable level. The overall and detail design of the facility is arranged so that the safety function of all components is identified and defined. In some cases there will be no safety critical function of a component. Usually there will be one or more safety functions applicable to different conditions. Amongst these conditions there may be the during- and post-earthquake conditions.

The process of determining the detail of the earthquake safety functional requirements is divided into five stages:

Stage 1. Identification of areas where seismic protection would be most beneficial.
Stage 2. Definition of those safety functions which would restrict the consequences following a seismic event.
Stage 3. Determine which of the safety functions should be achieved by design in order that the criteria are satisfied.
Stage 4. Identification of additional requirements for seismic events less frequent than 1 in 10,000 years.
Stage 5. Identification of additional requirements because of probabilistic criteria.

From the application of these stages a list of the functions required of buildings and plant and equipment is developed. The list is unique to each facility but to aid the description of the functional requirements categorisation of the requirements is employed. The categorisation captures the seismic performance requirements typically required of building structures and of plant and equipment. The categorisations are detailed below.

In addition to identifying and listing the seismic functional requirements, the facility will also be arranged so as to reduce the extent and number of items requiring seismic qualification. This is carried out through segregation of plant and equipment that have seismic design requirements from those not having such a requirement.

4.1. Building structures seismic design categories

Building structures are placed into one of the following categories:

C1a Water retaining structures which are to remain elastic. Storage ponds are typically in this category.

C1b Water retaining structure with a ductile water retaining liner. Lined process cells are typical of this category. The lining may be provided to retain liquors that might spill from vessels which are not designed or guaranteed to retain liquors on the occurrence of a DBE.

C2 Deformation sensitive structure that is to remain elastic. This category would be employed for buildings holding mechanical equipment which need to maintain accurate alignment and which are required to have a post-earthquake function.

C3 Limited ductility is permitted with some deflection limits applied.

C4 Structure is to remain standing. Ductile capacity of the structure can be employed.

S Special restricted performance limits to meet functional requirements. This category may mix C1 — C4 and other requirements.

The seismic design categories applied to the structures that make up a large facility may differ in different parts of the facility. Wet process cells (normally of thick concrete wall construction with a stainless steel liner) typically have more onerous requirements, and commensurate categorisation, than the overbuilding over the process cells (of steel frame construction) holding nonessential inactive services and offices.

4.2. Plant and equipment seismic design categories

The functional requirements for plant and equipment during and following a design basis earthquake are numerous due to the wide variety and function of plant and equipment. The categorisations employed reflect this by being broad in scope. The categorisations used are:

S1 - Seismic qualification is to be carried out using shake table testing combined with functional tests. When this category is applied it is most frequently to electrical equipment such as switchgear, relay cabinets etc. for which it is difficult to prove the required functionality by other means.

S2 - The equipment is required to function during/after the DBE in a mode substantially as normal operation. Equipment which might be placed in this category are cooling coils of highly active storage tanks and related pipework.

S3 - The functions required post earthquake are reduced in comparison with normal operation. Vessels required to retain liquor but not required for continued operation, and cranes that must be able to retain and safely release loads in a controlled way, are examples of cases where this category would be applied.

S4 - Plant and equipment in this category must not impair the required functioning of S1, S2 and S3 equipment. S4 equipment is not required for operation after the DBE and may be in any condition after the seismic event, including partial collapse, provided it does not have an impact on higher rated equipment.

The categorisation of equipment is listed in the Pre-Commencement Safety Reports (PCSR) so that it is clear at an early stage in a project what will be required in the design of plant and equipment. In many cases the categorisation alone does not provide sufficient detail of the seismic requirements. Supplementary information is supplied to designers in the PCSR or supporting documentation to detail the functions required. Some projects group this supplementary information and produce sub-category descriptions and labels (prefixed S1 — S4). These uniquely identify the required functions of groups of equipment which are to function in the same way.

5. ACCEPTANCE CRITERIA

As well as descriptions of the seismic functional requirements for building structures, plant and equipment, it is necessary to have acceptance criteria to judge whether the components will meet the functional requirements. Since a range of different functional requirements is used there is also a corresponding range of acceptance criteria. The acceptance criteria are arranged to match the functional requirements. More onerous criteria are used for the components requiring more onerous during- or post-earthquake functions.

The acceptance criteria are drawn from national and international standards and codes of practice wherever possible. Where considered necessary requirements additional to those of standards and codes are imposed to ensure that the performance on the occurrence of a DBE is satisfactory.

5.1. Building structures seismic design acceptance criteria

The acceptance criteria for building structures are laid out in a BNFL document that deals with the seismic design of structures. The principal codes referenced for concrete structures are:

1. USA Nuclear Pressure Vessel Codes ASME III Division 2, and
2. USA Concrete Structures Codes ACI-318.

For steel structures the codes employed are:

1. AISC, the USA stress limit code, and
2. the UK steel structures code BS5950.

In addition guidance documents such as NEHRP and FEMA (formerly ATC306) are extensively used. For the assessment of existing structures built before seismic design was required, particularly for unreinforced masonry, acceptance criteria developed from research material not currently incorporated into codes may be employed.

5.2. Plant and equipment seismic design acceptance criteria

The acceptance criteria used for seismic design of plant and equipment are based on applicable codes where possible. However some types of plant and equipment items are not included in codes (e.g. shield doors) whilst others have applicable codes only for normal loadings with no provisions to cover seismic loading. Broadly plant and equipment fall into six groups. The groups and the acceptance criteria for them are:

1. Mechanical equipment, e.g. cranes, shield doors, mechanical handling equipment, acceptance criteria based on UK crane design code BS2573.
2. Containment equipment, e.g. vessels, piping, gloveboxes, ventilation systems, acceptance criteria use USA Nuclear Pressure Vessel Code ASME III.
3. Electrical equipment, e.g. switchgear, monitoring equipment. Acceptance criteria based on functional tests and shake table testing are typically applied. These are based on IEEE 344.
4. Storage equipment, e.g. stillage stacks. Acceptance criteria appropriate to the required performance are specified. These may be as containment or mechanical equipment or

structural criteria depending on what aspect of performance is being considered. The overall performance of some storage systems has also been verified in shake table tests.

5. Rotating equipment, e.g. fan motor drives, generators. This group of equipment is mostly associated with electrical systems. The equipment is typically robust and substantial sources of vibration. Acceptance criteria for this group of equipment are based on the criteria used for mechanical equipment, but may also involve reference to the robust performance of such equipment in past earthquakes.
6. Distributed systems, e.g. piping, ventilation ducts, cable raceways, can feature prominently in the S4 seismic category since they are often to be found overhead to other more important equipment having more onerous seismic performance requirements. Acceptance criteria for anchorage and supports are of primary importance. The acceptance criteria used for supports are based on codes values for piping supports (as used for containment equipment) and anchor manufacturers rules for dynamic load capacity of anchorages. The acceptance criteria for the supported equipment itself are developed from those for piping and from screening rules as used in the USA for the seismic qualification of similar equipment in existing nuclear power plants.

For each type of equipment acceptance criteria appropriate for the seismic design categories listed in Sub-section 4.2 are employed. Where acceptance criteria from the ASME III code are used the different Service Levels described in the code are conveniently used for the various design categories. For example the S2 design category is usually assessed against Service Level B and the S3 category usually against Service Level D. When other codes are used, the acceptance criteria for S2 category equipment is usually allied to those for normal operation whilst S3 and S4 category criteria use those for out of service or occasional loads.

6. SEISMIC DESIGN LOADINGS

The seismic design of building structures, plant and equipment is carried out in respect of seismic loads from the design basis earthquake as described in Section 3. The form of the design loadings however is varied according to the component being analysed and the analysis procedure employed. The loading descriptions normally employed are:

1. Accelerations. These may be factored values from the free-field spectra or those obtained from time history or response spectrum analyses of building structures. Factored free-field values are only used for preliminary design. Acceleration values are seldom used as the primary design loading for building structures.
2. Response spectra are frequently used as the seismic loading description for buildings and equipment. As for accelerations, factored free-field spectra may be used for preliminary design. For equipment design ordinarily a sequence of preliminary spectra, approved spectra and revised spectra will be issued by the building designers for the design of equipment. These are normally obtained from time history analyses of the building structures. Broadening and lopping of spectral peaks may be applied. The revised spectra are obtained where conservatively calculated spectra are excessively onerous for equipment design or when local changes to the building structure or equipment invalidate the original spectra.
3. Relative displacements. Inter storey drift displacements are typically applied to column type vessels and pre-assembled modules which are restrained to building structures at a number of elevation levels. Occasionally relative displacement type loading may also be

applied in the seismic design of mobile mechanical plant such as cranes and shield doors.

4. Time history loads. Time history loading is normally used in soil-structure interaction analyses. The loads developed in the building structures from these analyses may then be used for design of the building. For equipment time history type loads are either directly obtained from building structure analyses as displacement histories or are derived from response spectra. The spectrum derived (response spectrum compatible) histories are calculated where it is necessary to envelope a number of response spectra to reduce the number of equipment time history analyses to be performed. They may also be used to incorporate response spectrum peak broadening and lopping in the design loading for equipment.

In addition to these normally employed loading parameters, building structural modal properties may be used as part of the loading information in building/equipment coupled systems response spectrum analyses.

7. DESIGN METHODS

The methods employed to determine properties to compare against acceptance criteria are selected from those appropriate to the quantities to be derived and seismic loading information available for the analyses. The most frequently used methods have been available for many years. These are:

1. Static equivalent loads analysis with conservatively set acceleration loads.
2. Natural frequency assessment followed by static equivalent loads analysis. The natural frequency assessment allows a less over-conservative loading to be employed.
3. Single input response spectrum analysis. Three input spectra are used, one for each orthogonal direction, and all restraints in each separate direction have the same spectra applied.
4. Time history analysis. Inputs at different locations may differ. Both linear and non-linear analyses are performed.
5. Comparison to previously qualified plant and equipment. Where designs for a new facility are closely based upon those of equipment already qualified for recent new facilities assessment by comparison may be employed to reduce the usual considerable expense of seismic assessment and qualification.

Less frequently used are more recently developed methods such are coupled systems response spectrum analyses.

Hand calculations, electronic workbook calculations and finite element methods are all used as appropriate to the analysis technique.

In a few cases natural frequencies and normal modes will be determined by tests. However in recent years these tests have been only to confirm the values obtained by calculation.

Shake table testing is also used as a qualification method. These tests directly measure the properties to be compared against the acceptance criteria.

In general the designers and analysts for a building structure or equipment item are not compelled to use particular methods. They are generally free to use the methods which will lead them to an economical design, provided the method is also generally accepted. In some cases, particularly with plant and equipment, specific methods may be required in the specification for detailed design and supply contracts. A recent example of this was with the interconnected gloveboxes in the Sellafield Mixed Oxide Fuel Plant described below in Sub-section 8.1. For seismic qualification of cranes (in- and out- cell electric overhead travelling cranes, not dockside or vehicle mounted types) suppliers responsible for the detailed seismic design are encouraged to use simple static equivalent methods and take account of the load limiting effects that sliding of the crane or bogie can have. The crane designers occasionally choose to perform more elaborate dynamic analyses but these are thought to lead to no significant reduction in overall costs.

8. PLANT AND EQUIPMENT SEISMIC DESIGN PROCESS

The process typically used for the seismic design of plant an equipment is in six stages. These are:

1. Preliminary design. This stage initiates the design process and will establish the design sufficiently such that it will be feasible to execute the final detailed design without any major changes. This stage is often carried out in-house by BNFL design teams.
2. Preparation of Design Proposal Drawings (DPDs). These are drawn to reflect the preliminary design. Ordinarily the equipment envelope, i.e. the space that may be occupied by the equipment and major sub-assemblies, will be delineated but the detailed sizes of structural elements of the equipment will not be stated. The interface to other equipment or building structures are also detailed in the DPDs.
3. Preparation of equipment design specifications. The design specifications incorporate the seismic performance requirements from the PCSR and the DPDs. They also specify any assessment methods required to interface correctly with the seismic design of other systems.
4. Detailed design and supply contracts. Plant and equipment are normally supplied under detail design and supply contracts. The contractor is normally responsible for the whole design of the equipment. Contractors are usually allowed to sub-let the execution of (but not responsibility for) the seismic aspects of the design assessment to specialists. Many contractors with only occasional need for seismic design choose this option to ensure a sufficient level of expertise is employed.
5. Audit of suppliers seismic design calculations. This stage ensures that the seismic design qualification is achieved to an acceptable level and is adequately documented.
6. Walkdown of the as-constructed facility during commissioning. This stage is to confirm the seismic design and identify any feature which might adversely affect the seismic design which has not already been eliminated by the preceding stages. In particular the effects of interaction between equipment of different types and from different designers is reviewed. Procedures for walkdown and resolution of walkdown findings are incorporated into normal project procedures and documentation. Post-construction walkdown is considered further in Section 9.

As noted above the method to be employed by the detail designers as part of the overall seismic qualification process is sometimes included in the design specification. An example of a case where the design method was specified is given in the following Sub-section.

8.1. Seismic design method for interconnected glovebox systems

In developing the design procedure for suites of interconnected gloveboxes it was realised at an early stage that the contract strategy of having separate design and build contracts for each glovebox would have a significant impact on the approach to seismic design since the glovebox suites are physically connected. This is because each glovebox of a suite would dynamically interact in an earthquake. The design loads for each glovebox are then dependent on the properties of all parts of the suite. To determine these loads co-ordination would be required during design between the design contractors for all parts of a suite and large scale dynamic analyses carried out. This would make it difficult to freeze the design. Accordingly a procedure was developed which would involve no interaction between design contractors yet still allow all the gloveboxes to be connected together.

The procedure adopted was composed of the following elements:

1. the gloveboxes or tunnels of each contractor were to be made effectively seismically rigid by requiring their natural frequencies when analysed separately to be above 20 Hz;
2. between each glovebox or glovebox and transfer tunnel (a type of glovebox which is solely used to connect gloveboxes and transfer material from one glovebox to another) which were part of separate design contracts, a component termed a Seismic Displacement Absorber (SDA) would be fitted;
3. the gloveboxes and transfer tunnels would have design loads applied to them at the attachment positions of the SDAs equal to the load at the maximum rated displacement of the SDAs;
4. in addition to the SDA load, each component of the glovebox suite would be designed against earthquake static equivalent loads.

The first element of the procedure was possible since the building structure frequencies were well below the 20 Hz value. The requirement necessarily leads to modest displacements at any connection between components of a suite. It also allows the last element of the procedure to be applied and reduces the design loads from those that would be required if the suite components were allowed to have lower frequencies.

In the second part of the procedure SDAs are required. SDAs are stainless steel bellows units which allow relative movement between the ends of the units with only modest loads being developed. Since only small loads would be developed when displaced by the attached glovebox or tunnel, and the SDAs have a relatively low mass, the glovebox or tunnel frequency calculations of the first part of the procedure could safely omit any interaction with adjacent suite components. The SDAs were designed to accommodate movement more than double that to be expected from any seismically rigid glovebox or transfer tunnel. SDAs to a limited range of sizes were specially manufactured at little cost in relation to the savings obtained in facilitating the design process. Tests were carried out on the SDAs to confirm they correctly functioned under earthquake conditions.

The third element of the procedure allows the full displacement range of the SDAs to be used without compromising the integrity of the glovebox suite components. It avoids having to determine actual relative displacements between components supplied by different contractors and communicating the resultant loads between contractors.

As well as reducing design loads, the fourth element of the procedure allows simpler design analysis and assessment that experience shows has a greater likelihood of being efficiently and rapidly executed than more elaborate dynamic methods. It also is reasonably familiar to the Design and Supply contractors responsible for executing the design, calculations and manufacture of equipment.

8.2. Plant & equipment seismic design guidance procedures

Some types of plant and equipment feature prominently in the lists of equipment requiring seismic qualification and are not adequately covered by seismic design codes. BNFL therefore decided to prepare some guidance documents for designers and analysts to aid efficient and consistent seismic design and assessment.

Guidance documents prepared so far include those for cranes, gloveboxes and shield doors. In addition, procedures for shake table testing, mainly of electrical equipment, have been prepared and routinely employed. These are extended for new equipment types as required. The guidance documents are incorporated or referenced by equipment design and supply contracts as appropriate. A further document on anchorage of equipment is in preparation to speed anchorage design and ensure consistency across a number of projects.

As well as design guidance documents, some similar papers on the assessment of existing plant and equipment have been prepared. A procedure for assessing cable raceways is a recent example.

9. POST-CONSTRUCTION WALKDOWN

Post-construction walkdowns are carried out during the commissioning of new seismically qualified facilities. The walkdowns are focused on the plant and equipment in the facility but regard is also taken of potential adverse effects that might arise from the seismic performance of building structure components.

The walkdowns have common features to those employed in the USA for the seismic qualification of existing facilities, particularly in the use of information on the seismic performance of industrial equipment, the use of screening rules and investigation of outliers. However the purposes of the post-construction walkdowns differ and are:

1. That there will be no features in the plant and equipment which might adversely affect the required seismic performance.
2. Confirm the seismic design of the plant and equipment. When combined with the audit of the seismic design calculations (noted in Section 8) this ensures no changes are made that are not incorporated into design calculations.
3. Confirm anchorage installation. This has the same effect as item 2 above but is highlighted since adequate anchorage is often considered to be a prerequisite to good seismic performance.
4. Identify potential interactions between equipment. This aspect is sometimes difficult to capture during the equipment design stage, particularly if site run services are present that only appear in schematic form on design drawings.
5. Identify potential interactions between building structure components and plant and equipment.
6. Identify components which have been the cause of damage in past earthquakes.

Where features are found which might adversely affect the required seismic performance, then plant modifications are made. The procedures employed to implement modifications are the same as used for resolution of other types of plant commissioning modifications.

9.1. Post-construction walkdown findings

International experts in assessment of equipment seismic performance have noted that both new and old industrial facilities practically without exception have features which would perform poorly in an earthquake. Accordingly it was expected that post-construction walkdowns would detect some features in the new facilities prior to completion of commissioning that would not give entirely satisfactory seismic performance. The features which have been recorded as commissioning faults found by walkdowns, and the means used for their rectification are noted below.

Installation faults:

- In a few cases packing plates between the equipment baseplates or feet of equipment stands and encast floor plates were inserted between hold down bolts and were of smaller plan area than the baseplates or feet. To avoid excessive bending loads on the hold down bolts the packing plates were replaced with plates of a size to fit the baseplates or feet and having holes sized to fit the hold down bolts.
- A few support bracing members for transfer tunnels had not been fitted due to clashes with the routing of cables for equipment in the tunnels. Where a tunnel lacked a support bracing member there was only ever one brace not fitted and the member was available for fitting. The members were fitted to give the designed arrangement or more calculations performed to show that the particular member omitted could be dispensed with.
- A few nuts/bolts on the support members for transfer tunnels had not been tightened and a few others had not been fully installed due to restricted clearances.

Interaction faults:

- Some site run equipment which has not been specifically seismically qualified and which had potential to fall was observed overhead to the seismically rated equipment. Overhead equipment was restricted to ventilation ducts (both rectangular and circular cross-section), pipework and cable raceways. In plant areas where gloveboxes and transfer tunnels predominated, due to the very rugged construction of these plant items, the low mass of overhead equipment and short distances they could potentially fall, often lead to assessments that no significant damage could result. In other areas, and for a few instances in the glovebox areas, further consideration of the overhead equipment was needed. A variety of means were employed to arrive at a final situation in which no significant damage could result. These ranged through:

a) detailed calculations to show that no collapse of the overhead equipment occurs;
b) strengthening of the overhead equipment for the same result;
c) providing additional support to prevent overhead equipment falling sufficiently far to impact the seismically rated equipment;

d) tests on the vulnerable components of the seismically rated equipment that showed that they can withstand the impact from the collapse of overhead equipment, should collapse occur; and
e) protecting vulnerable components from impacts.

As well as the immediate correction of the faults observed, the walkdown findings are fed back to further improve and refine the procedures for design and installation so that fewer faults are found in walkdowns as current and other projects proceed. The post-construction walkdowns have further enhanced the confidence obtained by the design and seismic qualification process that plant and equipment will satisfy their required seismic safety functions.

10. CONCLUSION

This paper has reviewed the procedures and methods employed by BNFL to ensure that its facilities in the UK will not present any safety risk on the occurrence of a very rare seismic event. The design basis earthquake employed in the design of new facilities and in the assessment of existing facilities has been described. The key feature of the procedures, and fundamental to economical seismic qualification, are a) to clearly identify the seismic safety functions required of building structures, plant and equipment early in the design process, b) physically segregate as far as practicable equipment requiring and not requiring seismic qualification to minimise the number of items requiring qualification, c) to use a range of acceptance criteria matched to the required seismic performances and d) to use a corresponding range of design methods that require no more elaborate calculations than necessary. It is also recommended that post-construction walkdown during facility commissioning is desirable to ensure that no feature could impair the seismic required safety functions.

SEISMIC DESIGN CONSIDERATIONS FOR NUCLEAR FUEL CYCLE FACILITIES

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Abstract

During the last few decades, there have been considerable advances in the field of a seismic design of nuclear structures and components housed inside a Nuclear power Plant (NPP). The seismic design and qualification of theses systems and components are carried out through the use of well proven and established theoretical as well as experimental means. Many of the related research works pertaining to these methods are available in the published literature, codes, guides etc. Contrary to this, there is very little information available with regards to the seismic design aspects of the nuclear fuel cycle facilities. This is probably on account of the little importance attached to these facilities from the point of view of seismic loading. In reality, some of these facilities handle a large inventory of radioactive materials and, therefore, these facilities must survive during a seismic event without giving rise to any sort of undue radiological risk to the plant personnel and the public at large. Presented herein in this paper are the seismic design considerations which are adopted for the design of nuclear fuel cycle facilities in India.

1. INTRODUCTION

The term Nuclear Fuel Cycle Facilities essentially refers to those facilities meant for fuel fabrication, spent fuel storage, fuel reprocessing and the waste management. The primary objective of earthquake resistant design of these facilities is to prevent any such damage to the structures and equipment that could lead to significant exposures to plant personnel or members of the public. Invariably, the need for the earthquake resistant design of these facilities depends on the probability of seismic events and on the consequences of such events if no earthquake resistant design features were applied. Therefore, seismic issues are given due attention right from the stage of site selection for these facilities. The civil structures which house the various systems, components and piping for such facilities are usually massive in nature. This is because mostly the dimensions of various walls of the structures are based on radiation shielding considerations. Seismic design of these structures along with the various components and piping is normally carried out for an earthquake potential which is consistent with the consequences of their failure and their intended design life. A brief account of the various design features provided in the design of these facilities to cater for the seismic loading is given in this paper.

2. VARIOUS DESIGN SAFETY FEATURES

The design features relevant to the nuclear fuel cycle facilities are mostly governed by the various process features to which they are supposed to cater for. Whereas during the planning of conventional chemical engineering installations, emphasis is laid on the planning of processing equipment, during the planning of installations for the nuclear fuel cycle facilities, it is also necessary and equally important to consider simultaneously the planning for processing equipment, the maintenance strategy and the building design. This is due to the fact that once these facilities start operating, it is difficult to attend to any kind of modification or repair works on the plant systems and components on account of high radiation fields.

The buildings housing these facilities are of increased importance because they serve the purpose of protective function. They serve as shielding against the radiations emanating from the various processes and also act as protective barriers to guard against the external events such as the occurrence of an earthquake. The building structures for these facilities are normally complicated structures which have to cater for the various requirements from the process side. Amongst these requirements, the consideration of criteria for the maintenance of the process equipment in order to achieve a high plant availability and reduced radiation exposure during repair and maintenance works plays a major role. In addition, the layout of process equipment in accordance with the process-technical and radioactivity-aspects also leads to a complicated design of the civil structure. The component layout is invariably carried out in such a way that the components which are subjected to mechanical wear or to ageing when exposed to the radioactive environment are disposed at the front end of the buildings so that the components can be replaced under remote control with the help of suitable manipulators.

3. SAFETY AND SEISMIC CLASSIFICATION

3.1. Fuel reprocessing facilities

The fuel reprocessing facilities involve various chemical processes for the seperation of desired elements from the spent fuel received from a reactor. The fuel is normally kept in the spent fuel storage bay for a definite period of time before it is taken for reprocessing. The information on design codes, standards and the other associated literature related to such facilities is scanty in nature. As such, no single document contains comprehensive guidelines for the safety and seismic classification of these facilities. In order to arrive at a rational design basis for these facilities, the authors have tried to compare these facilities with a reactor system based on the available information in the literature. The reprocessing process has certain important features, described below, which determine the safety and seismic classification of the various systems and components:

(1) In a nuclear reactor system, the large source of stored energy requires that the safety systems controls respond rapidly to maintain safe conditions during all the operational states and also during all the abnormal events. There being no such large source of stored energy in nuclear fuel reprocessing facilities, the safety system design can take advantage of the relatively longer time periods generally required for the development of conditions hazardous to the safety of plant personnel and public.

(2) The fuel reprocessing plants do not have the high temperatures and pressures that are associated with the power reactors. Mostly, the process conditions are such that the pressure is close to atmospheric and the temperature hardly reaches around 100^0C. In a power reactor, most of the radioactive materials are encapsulated in the fuel assemblies and most of the activity in the fuel is due to short-lived radionuclides. However, in case of fuel reprocessing plants the activity due to these short-lived radionuclides is absent because of their decay in the spent fuel storage bay before the fuel is transferred for reprocessing. The radioactive materials are released from the fuel matrix during the chemical processing of the spent fuel.

(3) The reprocessing plants are designed to cater for the reprocessing needs from multiple number of reactors. However, the amount of fuel handled at one particular time of the processing may not be as large as in a reactor system and thus, the associated radiation risk with a reprocessing facility would be much smaller as compared to a reactor system.

(4) The fuel reprocessing plants are designed with multiple confinement barriers for the prevention of any unwanted spread of radioactive materials. This, thus, helps in containing the radioactivity even if there is a failure of mechanical system or components. Also, the material handled or processed is retained, contained and confined within known bounds in a reprocessing facility for the reasons of accountability and also to minimise the spread of radioactive contamination.
(5) The design of various components and systems assumes a great importance in case of fuel reprocessing plants in terms of their shape and size so as to avoid the formation of any critical mass at any stage. This needs to be ensured even in case of unanticipated failure of the systems or components. Therefore, integrity of various systems and components assumes a greater significance in this regard.
(6) These facilities shall be designed for all the natural phenomena and postulated accidents as stipulated in 10 CFR 50 Appendix A — Criteria 2 and 4 excluding those accidents which are pertinent to nuclear reactors only [1]. The criterion 2 is regarding design bases for protection against natural phenomena such as earthquakes, wind loading etc. whereas criterion 4 is related to the enviornmental and dynamic effects design bases [2].
(7) The chemical corrosion and erosion conditions encountered in these processes tend to be extremely severe, placing a great emphasis on design for the containment integrity.

The above features are peculiar to these facilities. Based on these, it is recommended that the fuel reprocessing facilities shall be classified as Safety Class-4 facilities and that the design of various structures, systems and components pertaining to these facilities shall be performed using ASME B & PV Code, Section VIII. This is because fuel reprocessing is a chemical process and does not involve any sort of nuclear reaction as in case of a reactor system and, therefore, it does not require the components to be classified as Safety Class-1, 2 or 3 since none of the components in a reprocessing plant perform the safety functions which are pertinent to these safety classes. Use of Safety Class-4 for their design is consistent with the requirement of safety function 'n' for this class of systems and components which is as given below:

- For components performing safety function 'n' , safety class-4 is assigned such that if they failed, would not result in the exposure of public or site personnel in excess of the prescribed limits.

This is achieved in case of these facilities by the use of multiple barriers to prevent the spread of radioactivity. This is also in conformance with the opinion expressed in ANSI N46.2.11-1977, wherein it is mentioned that the ASME Section III design rules for Safety Class-1,2 and 3 components are applicable to nuclear reactor systems and do not directly apply to chemical plants such as the reprocessing facilities [3]. As such even the provisions of ASME B & PV Code, Section VIII are also not applicable for the low pressure and temperature conditions encountered in a fuel reprocessing plant. This is because the provisions of ASME Code Section VIII are not mandatory for such low pressure and temperature conditions. However, as a minimum, for meeting the requirements of sound engineering design, fabrication and inspection, the use of ASME code is recommended. The design of various systems and components is carried out for a set of artificial pressure and temperature conditions along with the other loadings so that the provisions of the code will apply for their design. This is because the main considerations in the design are equipment operability and long-term integrity which are duly met by proper material selection and the use of large corrosion allowances consistent with the service. As far as the civil structures which house these facilities are concerned, the design is performed using the same standards as applicable to a nuclear reactor system. This is due to the fact that the civil structures form the ultimate barrier which helps in preventing and containing the radioactivity.

As far as the seismic classification of these facilities is concerned, because of the hazards associated with these facilities, it is necessary to categorise them as the high hazard facilities where the confinement of contents and public and environment protection are of paramount importance [4]. The performance goal for High Hazard facilities is to provide a very high degree of confidence that the hazardous materials are confined both during and after the occurrence of a natural phenomenon such as a seismic event. Maintaining the confinement of hazardous materials requires that the damage be limited within the confinement barriers. The High Hazard facilities handle substantial quantities of radioactive materials in the forms which may permit wide spread dispersion. Facilities in this category represent hazards with potential long term and wide spread effects and, hence, they are designed for the maximum potential earthquake for that site. This is on account of the reason that for the High Hazard facilities, a reasonable performance goal is an annual probability of exceedance of around 10^{-5} of damage beyond which hazardous material confinement is impaired. This performance goal approaches, at least for earthquake considerations, the performance goal for seismic induced core damage associated with the nuclear reactors. Therefore, it has been the practice to design the High Hazard facilities such as the fuel reprocessing facilities for the Safe Shutdown Earthquake (SSE).

3.2. Fuel fabrication facilities

The fuel fabrication facilities basically include the various processes which are required for the fabrication of fuel sub-assemblies which are used for the nuclear reactors. The facilities which usually handle the normal uranium oxide fuel or the natural uranium do not pose much of a safety concern as the amount of activity handled is very small. These facilities are, therefore, designed using ASME Code Section VIII or the other similar relevant codes and standards. For their design to cater for the seismic loading in India, use of normal building code such as IS-1893:1984 is made wherein a higher value of Importance Factor is usually adopted for defining the design basis earthquake motion. Other facilities which are meant for the fabrication of either enriched uranium or plutonium, need a greater level of safety as compared to the natural uranium fabrication facilities. These facilities shall have multiple barriers for the prevention of spread of radioactivity and shall be designed using procedures similar to those given in ASME Code Section VIII. However, their seismic design shall be carried out using the earthquake of the level of Safe Shutdown Earthquake (SSE). These facilities also fall under the category of High Hazard facilities because the spread of powdered fuel in these facilities is quite dangerous for both the public as well as the plant personnel. However, the fuel enrichment facilities which handle relatively smaller amounts of radioactive materials with less consequences are categorised as Medium Hazard Facilities. Such facilities are designed for an earthquake of the level of Operating Basis Earthquake (OBE). The annual hazard exceedance probability for such medium hazard facilities is of the order of 10^{-3}.

3.3. Spent fuel storage facilities

The irradiated fuel after coming out of the power reactors is normally stored in spent fuel storage bays. This is done to reduce the decay heat and activity levels in the fuel before it can be sent for reprocessing or disposed off. Spent fuel storage facilities usually contain a pool of water under which the spent fuel is stored in the spent fuel storage racks. The water in the spent fuel pool is continuously circulated through a system consisting of the associated piping, pumps, valves and heat exchangers. The spent fuel storage facility in a nuclear power plant or in a fuel reprocessing facility is classified as a Safety Class-3 facility because this safety class incorporates all the safety functions associated with maintaining sub-criticality of the fuel stored outside the reactor coolant system and with the removal of decay heat from irradiated fuel stored outside the reactor coolant system. As far as their seismic design is concerned, these facilities are designed for an earthquake of the level of SSE [5,6]. The Project Design Safety Committee (PDSC) in India recommends similar guidelines for the design of these facilities.

However, the design rules for the independent spent fuel storage installations are somewhat different. The salient features which are peculiar to these installations are as follows:

(1) Such facilities are independent of both a nuclear power plant and a reprocessing facility and, therefore, the consequent risk associated with these facilities is much less.
(2) The storage of spent fuel in these installations is a low hazard potential activity. This is because very little of the radioactivity present is available in a dispersible form and there is no mechanism present to cause the release of radioactive materials in significant quantities from the installation.
(3) A risk study performed for these facilities utilising the conceptual design approach, site selection criteria, various design basis events etc. indicates that the radiological risk associated with these installations is 2 to 3 orders of magnitude lower than that of a nuclear power plant [7,8].
(4) The independent spent fuel storage installations are designed to resist an earthquake which has a mean recurrence interval of 500 years or in other words, the probability of occurrence for the design basis seismic event is taken as 2×10^{-3} per year.

These features are introduced in their design by performing their design using the standards similar to ASME Code Section VIII. The seismic design for these installations is carried out for an OBE level of earthquake.

3.4. Nuclear waste management facilities

The nuclear waste generated in reactor systems is usually handled by the liquid radioactive waste processing system which is generally attached with the nuclear reactor. This system is not a safety system nor does it contain the components that need to be safety class. The activity levels in these systems are much lower than that in a spent fuel storage bay or in a reprocessing plant. This system is, therefore, classified as a Safety Class-4 system with its design intent being met by performing the design in accordance with the provisions of ASME Code Section VIII or other similar relevant standards. The seismic design of various components and piping is performed for the OBE level of earthquake. The building housing these systems too is designed to cater for the OBE loading [9].

The nuclear waste generated from a reactor system as well as from a fuel reprocessing plant is stored in huge waste storage tanks in a Waste Tanks Farm (WTF). The WTF usually has multiple number of waste tanks which cater for the various types of wastes generated depending on their activity levels. Since the WTF is a store house of huge quantities of radioactive materials, it is classified as a Safety Class-3 system. This is because in performing the safety function ‘n’, the amount of radioactivity spread in the event of failure of such a system may lead to the radiation dose in excess of the prescribed limits [10].The civil structures housing these tanks are designed using the guidelines similar to a nuclear reactor system. This is because these structures are the ultimate barriers which contain the radioactivity and, thus, help in preventing the spread of radioactivity in the event of any failure on the process side. Similarly, for seismic design also, they are required to maintain their structural integrity in case of the maximum potential earthquake at the site; i.e. SSE. This requirement too comes from the large amount of the activity handled in these structures.

4. SEISMIC DESIGN METHODOLOGY

Currently, very few published guidelines are available regarding the seismic design of nuclear fuel cycle facilities. As compared to this, there is plenty of information available for the design of nuclear power plant components for seismic loading. At present, the designer is rather forced to adopt some of the conservative design and analytical techniques (meant for the design of reactor components and structures) for performing the seismic design of nuclear fuel cycle facilities. However, specific guidelines are needed to specify the methods of design and analysis for such facilities to ensure safe design of various critical systems and structures against the potential seismic hazards. It is worth mentioning here that there are many uncertainties associated with the earthquake response of real structures and systems. Much engineering judgement and experience is required to obtain meaningful results. This, therefore, many a time calls for a conservative seismic design for the critical systems and structures for these facilities. Hence the design of these items is carried out in such a manner that they remain functional even during and after the occurrence of a strong earthquake with very little or no structural damage The discussion here is restricted to the design of such critical equipment, systems and structures whose failure during a seismic event could result in a radioactive hazard to the public.

The nuclear fuel cycle facilities are generally designed with multiple confinement barriers for the control of radioactive materials. The structures, systems and components in these facilities are classified according to their function and the degree of integrity required for the plant safety in the following manner:

- seismic category I structures, systems and components are those whose failure could cause uncontrolled release of radioactive materials or those whose operation is required to effect and maintain a safe plant shutdown. Systems and equipment in this category are designed, constructed and inspected to withstand all the postulated loadings without loss of function. These are designed for the SSE loading.

- seismic category II structures and systems are those whose failure would not result in an uncontrolled release of radioactive materials and whose function is not required to effect and maintain a safe plant shutdown. These are invariably designed for OBE loading or any other appropriate level of seismic loading in few cases, which is consistent with the requirement of the structures and the systems.

- All the other systems and structures which are not covered above, are designed as codal structures and systems using the provisions of IS-1893:1984 which gives guidelines for the earthquake resistant design of industrial buildings.

For example, the control room of a fuel reprocessing facility is designed to be a seismic category I structure which is isolated from the process systems by remote instrument systems so that no transfer of radioactive materials into the control room can occur.

4.1. Design of civil structures

The civil structures in nuclear fuel cycle facilities may be either above the ground or below the ground. For example, most of the fuel fabrication facilities are housed in above ground structures, whereas the fuel reprocessing plant and the WTF are examples of below the ground structures. Seismic design of civil structures is performed using the latest available tools and techniques so as to ensure their containment function under all the postulated loading conditions. Analysis of civil structures is usually carried out using the finite element method (FEM) using the softwares such as COSMOS/M, NISA, ABAQUS etc. for the seismic category I and II structures. Soil-structure interaction is given due considerations while analysing the structures for earthquake loading. This is accomplished mostly through the use of frequency independent soil springs and dashpots in the analysis. It is mostly observed that the soil-structure interaction effects are quite predominant for these structures because of their stiff nature (as compared to the normal civil structures) wherein the structural stiffness becomes comparable to the soil or rock stiffness and thereby alters the dynamic characteristics of the structure. Seismic design of the structures is mostly carried out using the Response Spectrum Method (RSM). Time history analysis of these structures is also required for generating the Floor Time Histories (FTHs) which are required for the computation of Floor Response Spectra (FRS). These FRS are then used for the design of various systems, components and piping which get their support motions from the respective floor locations.

In the analysis of buildings having frame type of structures, use of lumped mass beam models is quite common wherein the masses are lumped at appropriate levels in the model which is consistent with the mass distribution in the real structure. The stiffness of the structure in these models is modelled with beam elements that reflect both the bending and the shear stiffness of the real structure. For the other complicated structures which have shear walls and frames, use of plate / shell elements is made with a lumped mass formulation. For both the kinds of models, FE mesh refinement is decided in such a way that all the modes up to 33 Hz are excited with a reasonable accuracy. This is very important in the case of these structures which generally have high mass participation in higher modes only because of the stiff nature of structures which are used for housing these facilities. Moreover, the model should be able to predict high frequencies correctly also because most of the equipment and piping systems lie in the high frequency zone of the applicable FRS. Design of all the critical structures is carried out using the same standards as those used for the design of civil structures for the nuclear plant.

4.2. Seismic design of systems and components

Design of various systems and components pertaining to these facilities for earthquake loading is performed by considering the seismic load in the faulted condition of design. The primary objective here is to maintain the structural integrity and the functional requirement for the safe plant shutdown under all the operational states. The structural integrity of various components is normally assessed by either an equivalent static method or the dynamic methods such as the time history method (THM) or response spectrum method (RSM). Seismic loading on the equipment is usually determined by analysing either a coupled building-equipment system or by conducting two separate analyses, one on the building structure and the other one on the equipment. If the mass and/or stiffness properties of the equipment are such that it could affect the overall building response or if the equipment is supported at multiple locations so as to affect the building response, then a coupled dynamic analysis is required. The criterion used to judge this interaction effect is as stipulated in USNRC SRP 3.7.2 [11].

The coupled analysis would require only one set of dynamic response calculations. However, the mathematical model required would be more complex. It may require many more degrees of freedom in order to include both the equipment and the support structure. This complex and larger model is necessary to capture accurately the response of the equipment. Such a model would not only require much more computer effort for its solution, but would also increase the likelihood of introducing errors in its solution by way of ill-conditioned mass and stiffness matrices. As compared to this, performing separate analyses on the building structure and the equipment is generally more practical. Although two sets of response calculations are required, the models are much more manageable and result in the use of much less manpower and computer time. The building analysis provides the seismic loading to the equipment in the form of FTH or FRS. Subsequently, the designer has the flexibility of performing either an equivalent static or dynamic analysis.

Analysis using the equivalent static method is normally carried out for the simple equipment which have predominantly higher mass excitation in their first fundamental mode of vibration or for the rigid equipment. In this method, the equipment is subjected to a static load equal to 1.5 times the acceleration corresponding to its first fundamental frequency which is obtained from the applicable FRS for the requisite damping value. In case the frequency of the equipment is not available, then the equivalent static analysis is performed using the static load equal to 1.5 times the acceleration corresponding to the peak in the applicable FRS. This analysis is performed separately for each direction of vibration and then the resulting response is combined using the method of Square Root of Sum of Squares (SRSS). However, the equivalent static method results in undesirable conservatism in many cases.

Amongst the dynamic methods of analysis, use of RSM is preferred over the use of THM on account of its simplicity, less time consumption and conservatism. In addition, the design of equipment using FRS caters to a number of earthquake time histories as compared to the use of an FTH which represents the floor motion for only a single earthquake. The FRS and FTH are peak broadened and smoothened before their use in design, to cater for the various uncertainties associated with the analysis such as the variation in material and soil properties, uncertainties in frequency calculations etc.[12]. While using RSM for the seismic design, the modal responses are combined using the 10 percent grouping method and the spatial responses are combined in an SRSS manner [13]. The inertial response of equipment is evaluated using the dynamic analysis only up to the modes which have their frequencies below the ZPA frequency in the applicable FRS. Subsequently, a missing mass correction is applied in an equivalent static manner for the mass which has not participated upto the floor ZPA frequency. Use of THM is made only in cases where non-linear effects are required to be modelled.

For the equipment which are supported at multiple locations, the seismic response is composed of two parts, namely the inertial response and the response due to Seismic Anchor Movement (SAM). The inertial response is usually found out either by using an envelope spectrum approach or by using the different motions at various support locations in the form of FRS or FTH. The envelope spectrum approach uses a spectrum which envelopes the spectra at various support locations. This approach results in a conservative prediction of seismic response in most of the cases except in few cases where the equipment is extending beyond its supports such as in case of an overhanging equipment. In such cases, the use of multiple response spectrum technique is recommended. The SAM response is computed in an equivalent static manner and it is combined with the inertial response in an absolute manner.

5. DESIGN OF A TYPICAL WASTE TANK FARM

There are various layouts for the Waste Tank Farm designs. A typical WTF unit houses four stainless steel tanks in a single rectangular concrete vault. The vault in turn sits on a massive raft. The entire structure is an underground structure located at a rocky site with a seismic potential of 0.2 g acceleration during an SSE event (Fig. 1). The tanks storing the waste are cylindrical vertical tanks with flat top and bottom heads with suitable stiffening arrangements. The dimensions of civil structure are decided based on radiation shielding considerations. The concrete vault is capped by a heavy top slab at the top. This top slab has a corridor upto a certain height for the movement of materials etc. The entire civil structures has been designed using ACI-349 standard, whereas the tanks are designed as per the provisions of ASME Code Section III Sub-section ND which is meant for the design of Safety Class-3 components.

The pressure inside the tanks is kept under sub-atmospheric conditions so as to avoid the leakage of radioactivity to the environment. The maximum vacuum for which the tanks have been designed is 6895 N/m^2 (1.0 psi). The waste in these tanks is continuously cooled by the

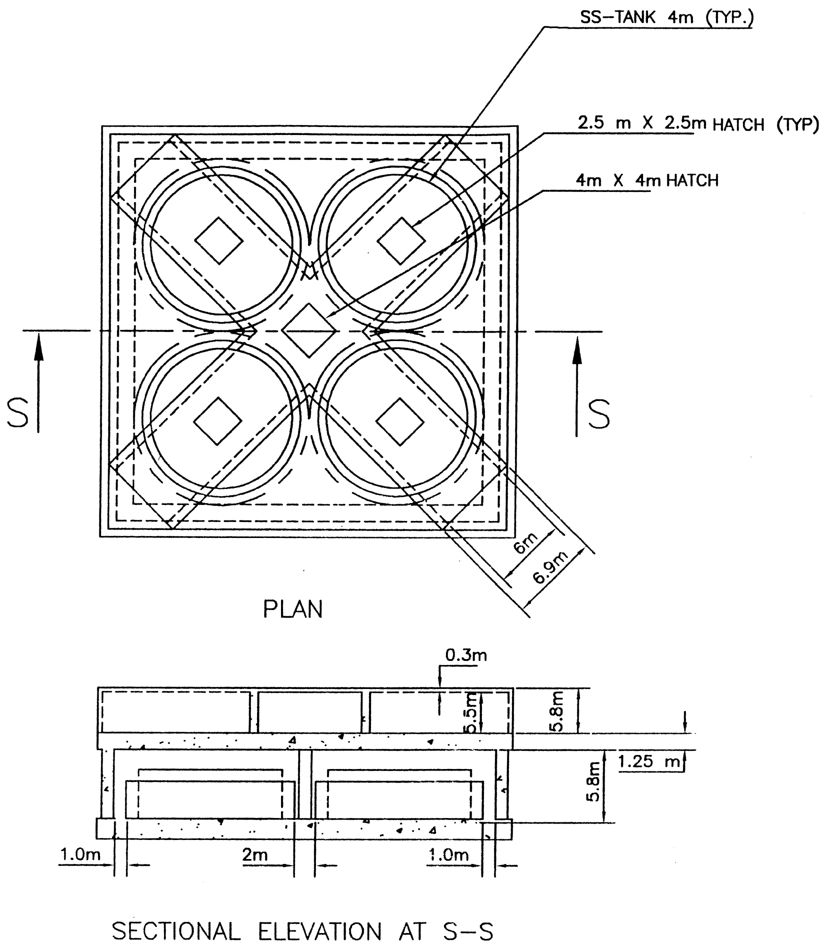


FIG. 1. Details of a typical WTF structure.

water flowing through the cooling coils which are present inside the tanks. The adequacy of the thickness of these tanks has been checked for both internal as well as external pressure as per the provisions of ASME Code Section III, Subsection ND since they are classified as Safety Class-3 components. The forces arising out of hydrodynamic effects during a seismic event have been evaluated as per the procedure given in the ASCE 4-86 standard [14]. The tank wall has been checked for the buckling under the combined effect of external pressure and the axial compressive loading due to the weight and earthquake loads as per the provisions of ASME Code Case N-284. In addition to this, separate stability checks have been performed to safeguard the tanks against the elephant foot buckling and the diamond buckling.

The concrete vault housing these tanks has been analysed for various postulated loadings using its finite element model (Fig. 2). This model is composed of plate/shell elements. The soil has been modelled in the form of frequency independent soil springs and dashpots at the base and the sides. Seismic response has been evaluated using the RSM wherein the modal responses have been combined using the 10 percent grouping method and the spatial responses have been combined using the SRSS method. The structure has been analysed for the first thirty modes of vibration upto 33 HZ. Figs. 3 and 4 respectively show the mode shapes for the first two modes of vibration. Similarly, the response due to the other loadings has also been computed. Design of the structure has been then carried out as per the provisions of ACI-349 standard.

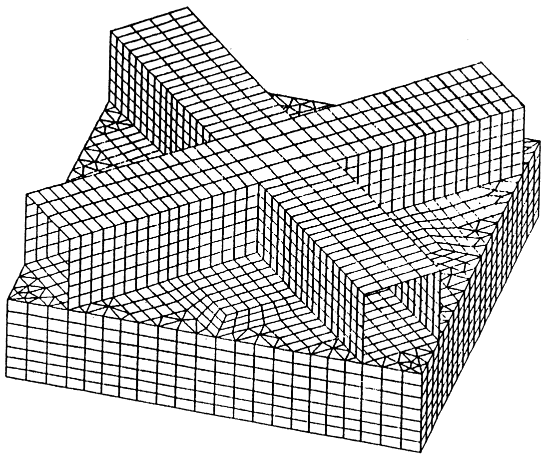


FIG. 2. FEM model of civil structure.

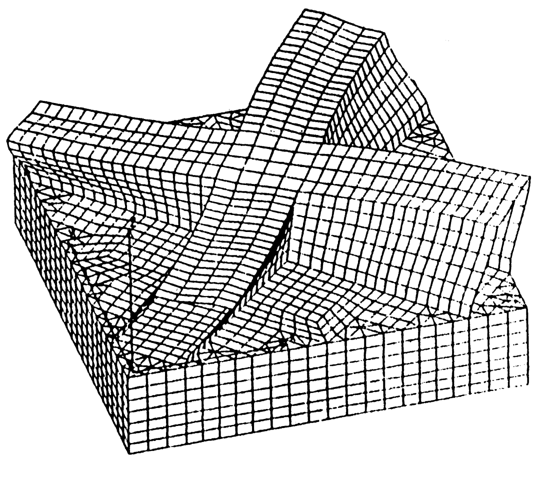
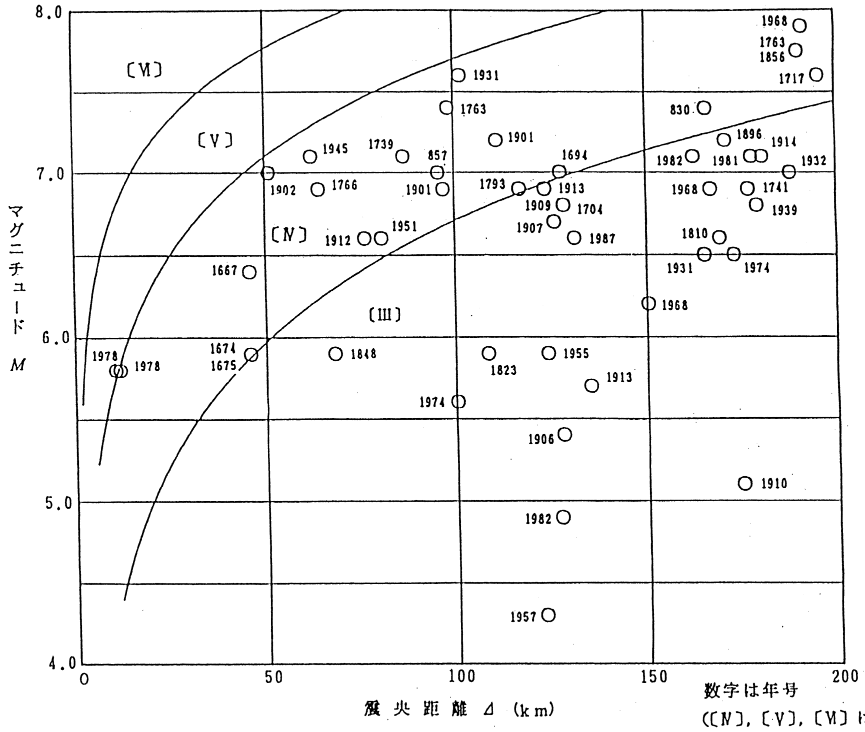


FIG. 3. First mode vibration of WTF structure (f = 11 Hz).



	年 月 日 地名または 地 震 名	基盤速度 (kine)
主な被害地震	1978. 5.16 青森県東岸	4.58
	1902. 1.30 三戸地方	4.30
	1978. 5.16 青森県東岸	4.22
	1945. 2.10 八戸北東沖	3.77
	1931. 3. 9 青森県南東方沖	3.75
	1968. 5.16 十勝沖地震	2.20

FIG. 4. Second mode vibration of WTF structure (f = 13 Hz).

ACKNOWLEDGEMENT

The authors gratefully acknowledge the useful contribution made by Shri G.J.Pandya, Shri Ved Singh and Shri K.V.Srinivas, Fuel Reprocessing Projects Division during the design and analysis of the typical Waste Tank Farm.

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SEISMIC DESIGN FOR NUCLEAR FUEL CYCLE FACILITIES IN JAPAN

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Abstract

Nuclear Safety Commission was determined the several safety examination guides for nuclear fuel cycle facilities. Each guide provides for the seismic design procedure. This paper is pointed out the outline of these seismic design guides and an application example of seismic design for reprocessing facility.

1. NUCLEAR SAFETY EXAMINATION GUIDES

Nuclear Safety Commission was determined the several safety examination guides for nuclear fuel cycle facilities;
Principal Guide for Safety Examination of Nuclear Fuel Facilities (Feb. 7, 1980)
Safety Evaluation Guide for Uranium Fuel Fabrication Facilities (Dec. 22, 1980)
Regulatory Guide for Licensing of Reprocessing Plants (Feb. 20, 1986)
Safety Evaluation Guide for MOX Fuel Fabrication Facilitie (Under investigation)

Related Guide;
Examination Guide for Seismic Design of Nuclear Power Reactor Facilities (Jul. 20, 1981)

2. RELATED ARITICLES FOR SEISMIC DESIGN FOR EACH GUIDE

Each Guide has almost same structure of articles to related the seismic design.

Article 1. Fundamental Site Condition;

(1) Natural Condition

1) Natural phenomena of earthquake, tsunami, landslide, subsidence, typhoon, high tide, floods, abnormal cold wave, heavy snowfall and others.
2) Ground condition, bearing capacity of soil, geological features of fault, topography, etc.
3) Meteorological condition of wind direction, wind speed, amount of rainfall, etc.
4) Water condition of river, ground water and hydraulic condition around the site.

(2) Social Condition;

1) Fire or explosion at the neighboring factories.
2) Flying objects by the accident of aircraft.
3) Utilization condition of water, product and distribution of food, distribution of population and others.

Article 13. Consideration for the Effect of Earthquake;

For important facility from viewpoint of safety,

1) Classification of importance for seismic design,
2) Refer the historical earthquake records and survey in and around the site,
3) Sufficiently safe design for the most suitable earthquake forces.

3. CLASSIFICATION OF IMPORTANCE FOR SEISMIC DESIGN (Table I and II)

Each Guide determined the classification of importance for seismic design based on the environmental effect by the possible radiation of atmospheric radioactivity during earthquake.

1) Uranium Fuel Fabrication Facilities;

Seismic design is done by only static earthquake force.

	Buildings and Structures	Equipment and Apparatus
Class 1	1.3 CI	1.5 CI + secondary check
Class 2	1.1 CI	1.4 CI
Class 3	1.0 CI	1.2 CI

Where, the value of the story shear coefficient CI is determined on the basic story shear coefficient (0.2) used in conventional buildings according to the Building Standard Code.

2) Reprocessing Facilities;

Design philosophy is almost same as the design method for Nuclear Power Plants. Static and dynamic earthquake forces are used for seismic design.

	Static force	Dynamic force
Class As		S2
Class A	3.0 CI	S1
Class B	1.5 CI	
Class C	1.0 CI	

Where, the basic earthquake motions, S1 and S2, are called the maximum design earthquake and the extreme design earthquake, respectively. They are defined at the rock outcrop of the site.

3) MOX Fuel Fabrication Facilities;

Design philosophy is almost same as the design method for Uranium Fuel Fabrication facilities, but the importance factor was different.

	Building and Structures	Equipment and Apparatus
First Class (S)	2.0 CI	2.4 CI
First Class	1.5 CI	1.8 CI
Second Class	1.3 CI	1.5 CI
Third Class	1.0 CI	1.2 CI

4. DESIGN EXAMPLE OF EARTHQUAKE MOTIONS FOR THE REPROCESSING PLANT

One private company is constructing the reprocessing plant at the northern part of Japan main-island (Shimokita, Aomori Prefecture). Figures 1–6 show the design examples of earthquake motions for the plant.

TABLE I. CLASSIFICATION OF THE IUMPORTANCE FOR THE NUCLEAR FUEL CYCLE FACILITIES (BUILDINGS AND STRUCTURES

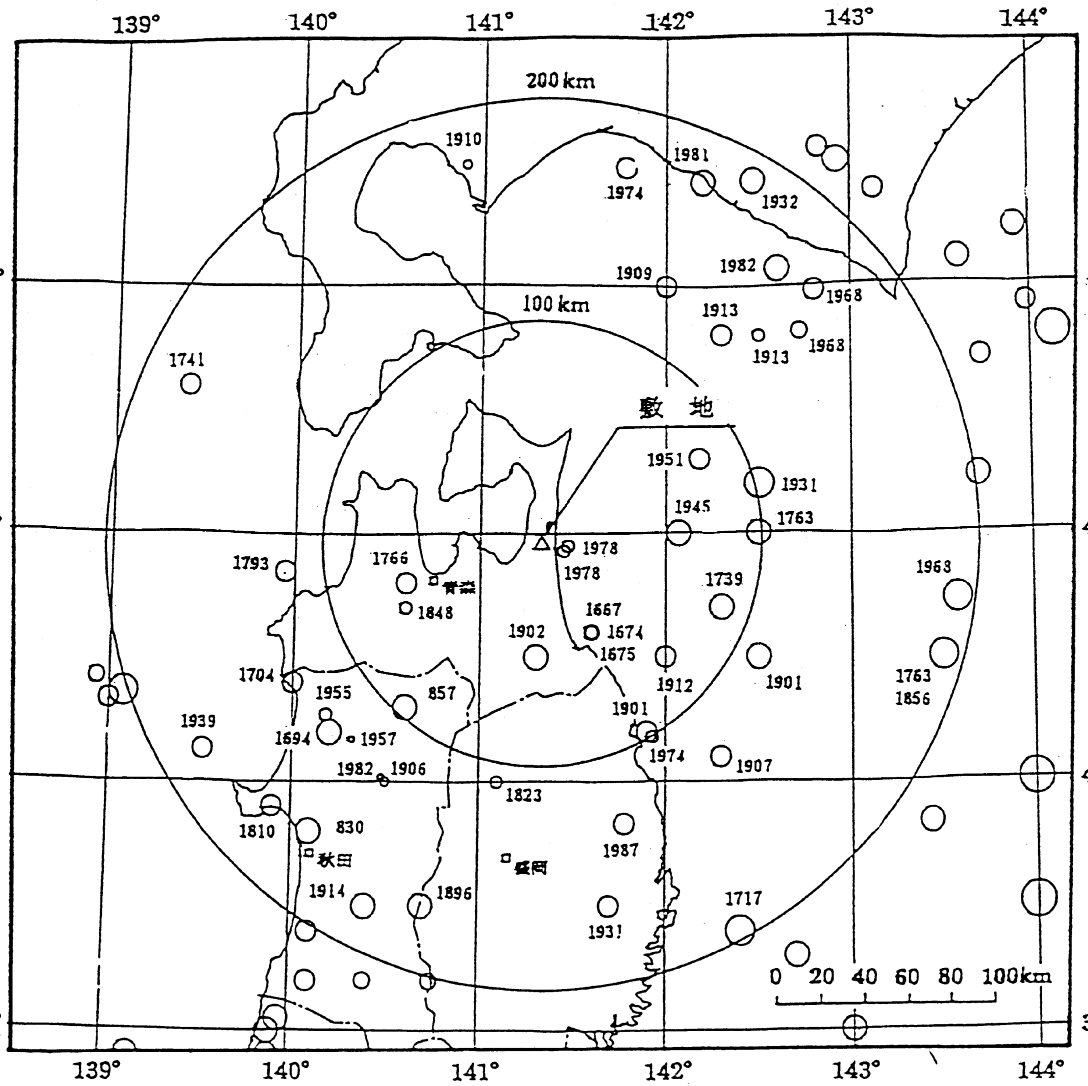
	Nuclear Power Plant, Reprocessing Plant			MOX Fuel Fabrication Plant		U-Fuel Fabrication Plant	
	Class	Static Seismic Force	Dynamic Seismic Force	Calss	Static Seismic Force	Class	Static Seismic Force
3.0	As	3.0 CI + CV	S2 (h) + 1/2 S2* (v)				
	A	3.0 CI + CV	S1 (h) + 1/2 S1* (v)				
2.0				1 (S)	2.0 CI		
1.5	B	1.5 CI		1	1.5 CI		
				2	1.3 CI	1	1.3 CI
						2	1.1 CI
1.0	C	1.0 CI		3	1.0 CI	3	1.0 CI

CI (story shear coefficient) : determind by consideration the dynamic characteristics of the structure, type of ground, etc. with 0.2 taken as the basic shear coefficient.
CV (vertical story seismic coefficient) : determined by considering the dynamic characteristics of the structure, type of ground, etc, eith 0.3 taken as the basic value
S2 (h) : horizontal seismic force of the extreme design earthquake motion
S1 (h) : horizontal seismic force of the mazimum design earthquake motion
S2* (v) : calculated by taken the maximum horizontal acceleration amplitude of the extreme design earthquake motion as the vertical seismic coefficient (1)
S1* (v) : calculated by taken the maximum horizontal acceleration amplitude of the maximum design earthquake motion as the vertical seismioc coefficient (1)
(1) : Both horizontal seismic force and vertical seismic force take place simultaneously combined in unfavorable directions. The vertical seismic force is considered to be constant in the height direction.

TABLE II. CLASSIFICATION OF THE IMPORTANCE FOR THE NUCLEAR FUEL CYCLE FACILITIES (EQUIPMENT AND APPARATUS)

	Nuclear Power Plant, Reprocessing Plant			MOX Fuel Fabrication Plant		U-Fuel Fabrication Plant	
	Class	Static Seismic Forc	Dynamic Seismic Force	Calss	Static Seismic Force	Class	Static Seismic Force
3.0	As	3.6 CI + CV	Kh (S2) + Kv (S2)				
	A	3.6 CI + CV	Kh (S1) + Kv (S1)				
2.0				1 (S)	2.4 CI		
1.5	B	1.8 CI	1/2 Kh (S1)	1	1.8 CI		
				2	1.5 CI	1	1.5 CI + Secondary check
1.0						2	1.4 CI
	C	1.2 CI		3	1.2 CI	3	1.2 CI

CI (story shear coefficient) : determind by consideration the dynamic characteristics of the structure, type of ground, etc. with 0.2 taken as the basic shear coefficient.
CV (vertical story seismic coefficient) : determined by considering the dynamic characteristics of the structure, type of ground, etc, eith 0.3 taken as the basic value
Kh (S2) : horizontal seismic coefficient calculated from the responce of the extreme design earthquake motion
Kh (S1) : horizontal seismic coefficient calculated from the responce of the mazimum design earthquake motion
Kv (S2) : vertical sesimic coefficient calculated by taken the maximum horizontal acceleration amplitude of the extreme design earthquake motion (1)
Kv (S1) : vertical seismic coefficient calculated by taken the maximum horizontal acceleration amplitude of the maximum design earthquake motion (1)
(1) : Both horizontal seismic force and vertical seismic force take place simultaneously combined in unfavorable directions. The vertical seismic force is considered to be constant in the height direction.



地震諸元は「宇佐美カタログ（1979）」による。ただし，1885年以降1980年までの地震については「宇津カタログ（1982）」，1981年以降の地震については「気象庁地震カタログ」による。

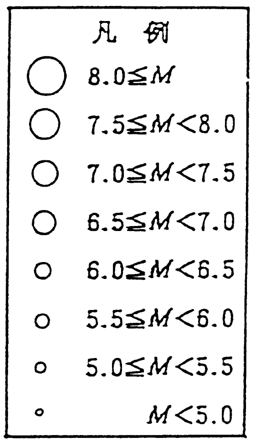
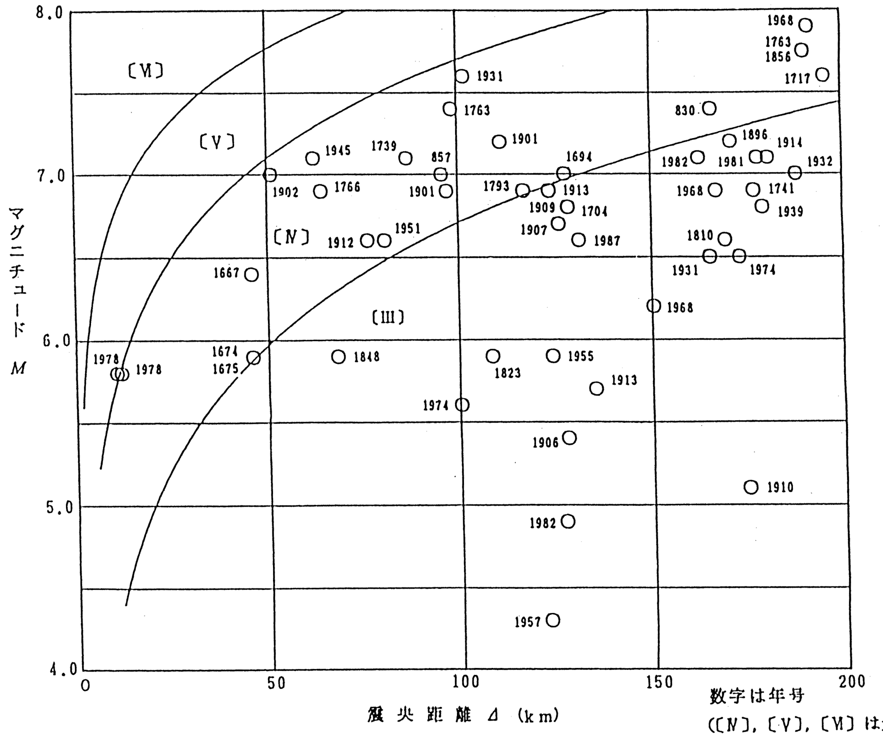


Fig. 1. The epicentre distribution of historical damaged earthquakes around the site.



	年　月　日 地名または 地　震　名	基盤速度 (kine)
主な被害地震	1978. 5.16 青森県東岸	4.58
	1902. 1.30 三戸地方	4.30
	1978. 5.16 青森県東岸	4.22
	1945. 2.10 八戸北東沖	3.77
	1931. 3. 9 青森県南東方沖	3.75
	1968. 5.16 十勝沖地震	2.20

数字は年号
(〔Ⅳ〕，〔Ⅴ〕，〔Ⅵ〕は気象庁震度階級で村松，勝又ほかによる。)

Fig. 2. The relation between magnitude and epicentre distance of the historical damaged earthquakes.

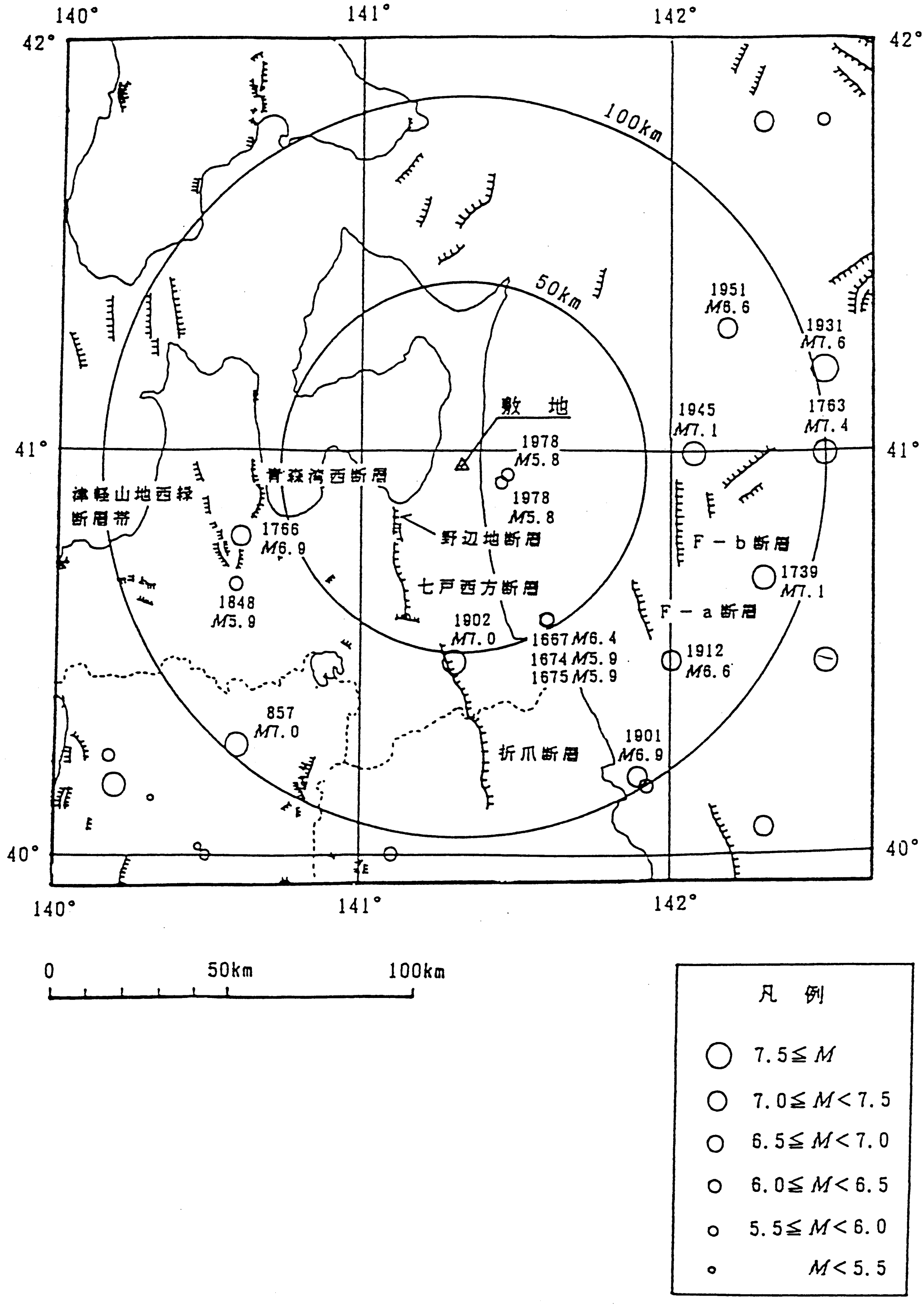


Fig. 3. The distribution of active faults and the epicentre of historical earthquakes.

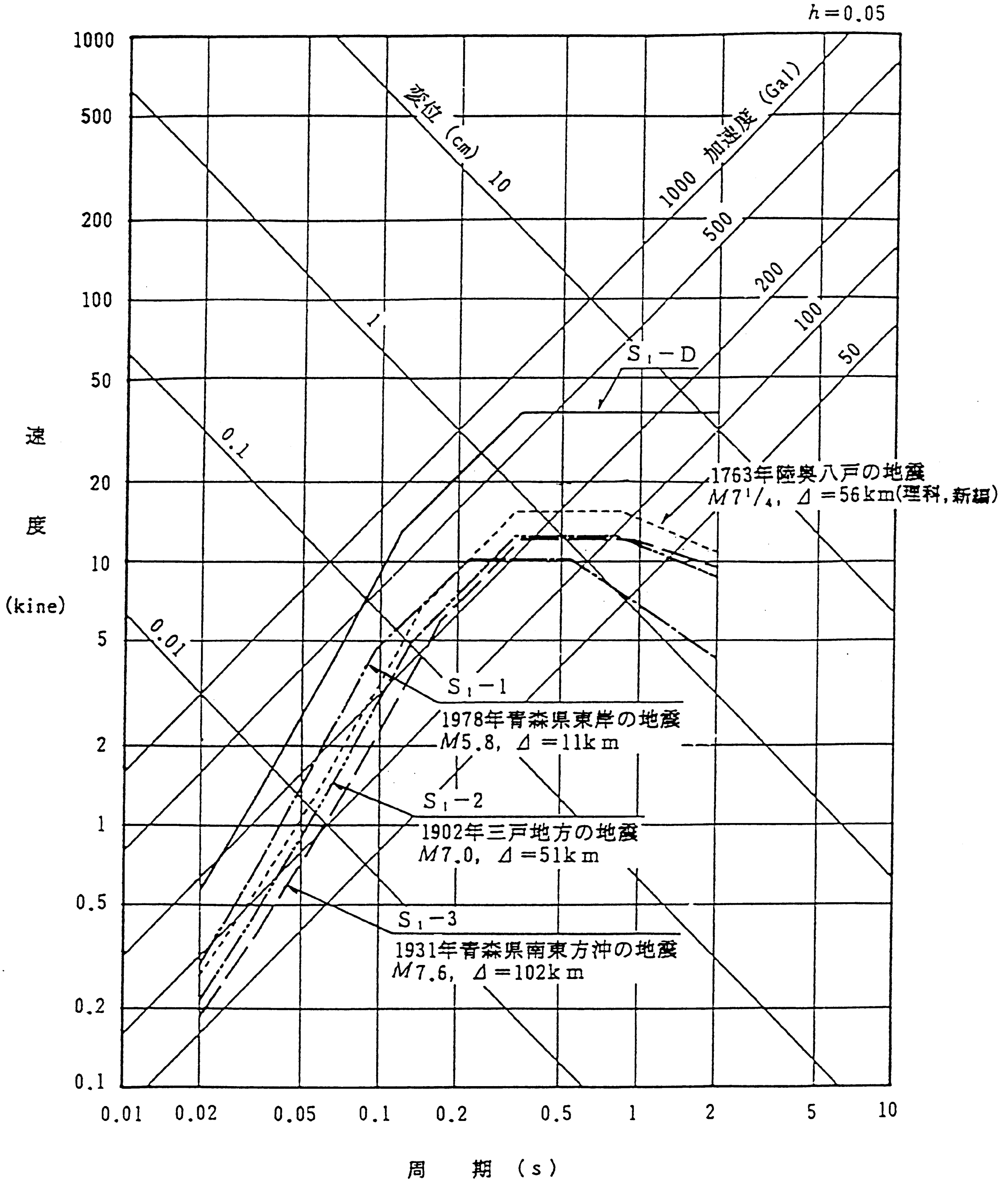


Fig. 4. The response spectra for the basic earthquake motion S1.

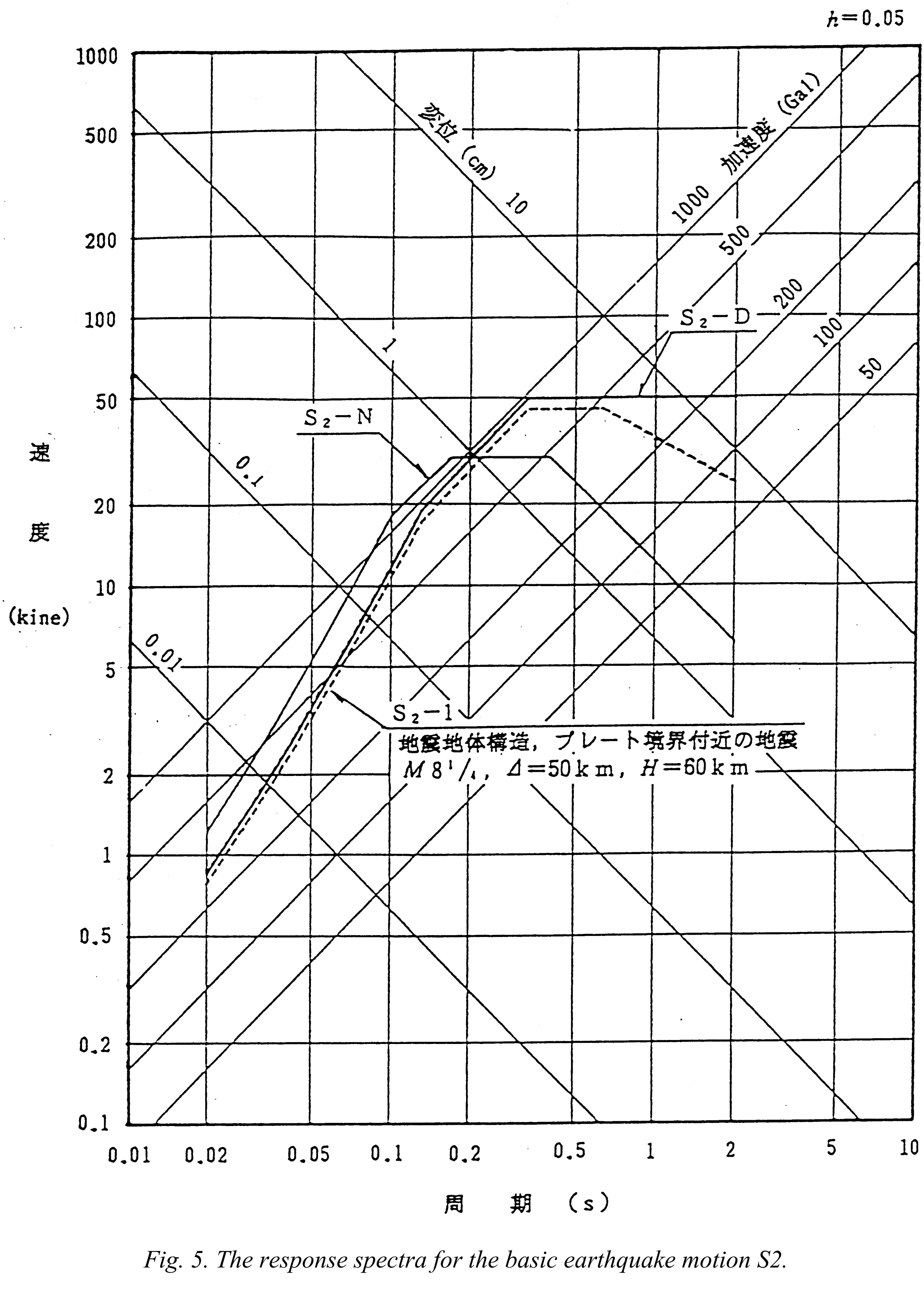


Fig. 5. The response spectra for the basic earthquake motion S2.

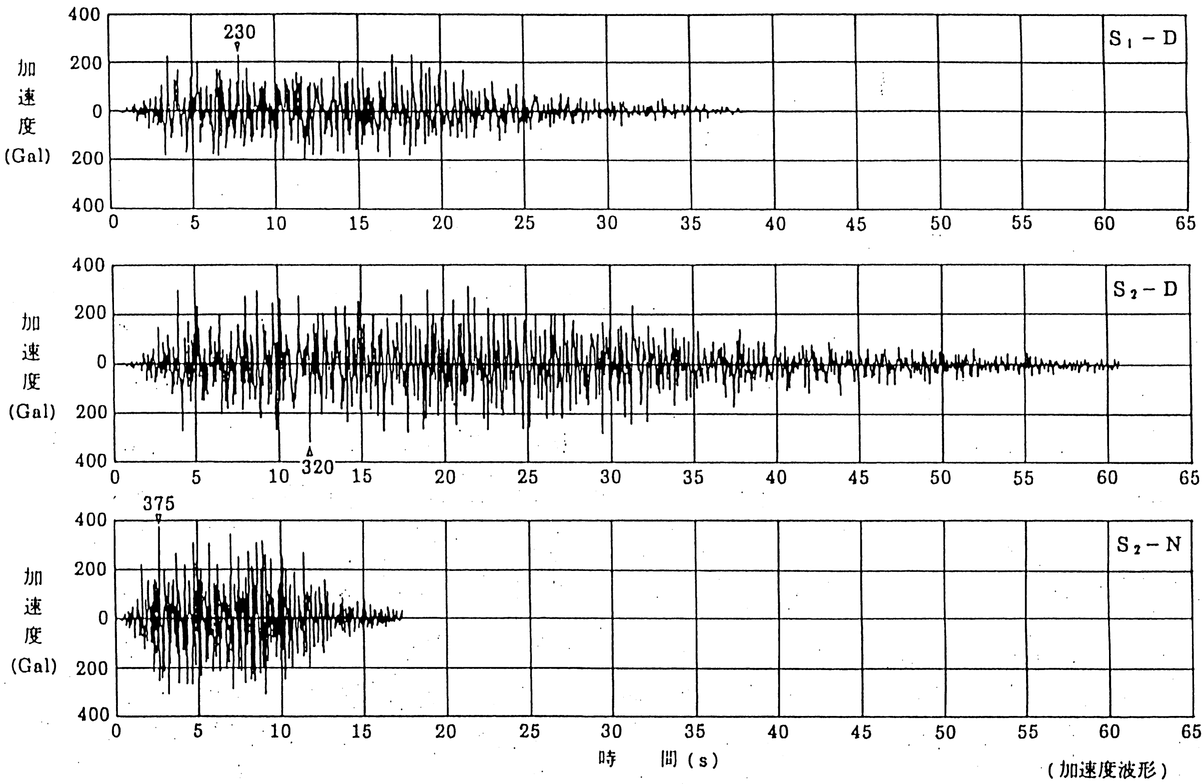


Fig. 6. The acceleration time-history of the artificial earthquake motions; S1, S2-D, S2-N.

REPORT ON THE SEISMIC SAFETY EXAMINATION OF NUCLEAR FACILITIES BASED ON THE 1995 HYOGOKEN-NANBU EARTHQUAKE

EXAMINATION COMMITTEE ON THE SEISMIC SAFETY
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Abstract

Just after the Hyogoken-Nanbu Earthquake occurred, Nuclear Safety Commission of Japan established a committee to examine the validity or related guidelines on the seismic design to be used for the safety examination. After the 8 monthes study, the committee confirmed that the validity of guidelines regulating the seismic design of nuclear facilities is not impaired even though on the basis of the Hyogoken-Nanbu earthquake. This report is the outline of the Committee's study results.

1. INTRODUCTION

From the standpoint of thoroughly confirming the seismic safety of nuclear facilities, Nuclear Safety Commission established an Examination Committee on the Seismic Safety of Nuclear Power Reactor Facilities (hereafter called Seismic Safety Examination Committee) based on the 1995 Hyogoken-Nanbu Earthquake on January 19, 1995, two days after the occurrence of the earthquake, in order to examine the validity or related guidelines on the seismic design to be used for the safety examination.

This report outlines the results of the examinations by the Seismic Safety Examination Committee. (The original report written by Japanese, and published on September 1995. Therefore, this English version is not official document.)

2. BASIC PRINCIPLE OF EXAMINATIONS AT THE SEISMIC SAFETY EXAMINATION COMMITTEE

In order to proceed the examinations and discussions at the Seismic Safety Examination Committee, it is important to collect as much information as possible from the Hyogoken-Nanbu Earthquake. Thus, tremendous number of reports or related documents prepared by the regulatory bodies, research institutes or academic societies were investigated as well as field investigations were conducted to obtain various information on the Hyogoken-Nanbu Earthquake, namely, earthquake parameters, source mechanism, displacement of fault, earthquake ground motion, and the damages to buildings and civil structures.

Based on the collected data and various information on the Hyogoken-Nanbu Earthquake, which has been understood comprehensively up to now, the items to be examined were selected so that the validity of related guidelines on the seismic design for nuclear facilities were to be examined in detail.

3. OVERVIEW ON THE RELATED GUIDELINES OF THE SEISMIC DESIGNS

(1) Related guidelines of seismic design

Nuclear facilities include light water reactors for power generation, fast breeder reactors, advanced thermal reactors, research reactors, nuclear fuel facilities (nuclear fuel recycling and reprocessing facilities) radioactive waste management facilities and radioactive waste repository.

With regard to the examination of the seismic design of these facilities, it is specified that the "Examination Guideline for Seismic Design of Nuclear Power Reactor Facilities" (hereinafter called "Seismic Design Examination Guideline") to be used or referenced.

(2) Overview on Seismic Design Examination Guideline

The Seismic Design Examination Guideline was systematically organized in 1978 by integrating the previously used conception on seismic design. Part of this guideline was revised in 1981.

Seismic Design Examination Guideline specifies the basic principle as follows: the nuclear power reactor facilities shall maintain its structural integrity against any hypothetic seismic force likely to occur at the site so that no earthquake brings about a major accident. Moreover, buildings and structures shall be, in principle, of rigid construction and the important buildings and structures shall be supported on bedrock. Also, it requires such severe seismic design as follows: the facilities of higher degree of importance shall be resistant to stronger seismic force than for usual commercial or industrial buildings and structures, the maximum design earthquake shall be evaluated based on historical earthquakes, active faults and vertical seismic force in addition to the horizontal seismic force shall be taken into consideration.

4. INFORMATION AND KNOWLEDGE OBTAINED ON THE 1995 HYOGOKEN-NANBU EARTHQUAKE

With regard to the Hyogoken-Nanbu Earthquake, many scientist, engineers and research institutes have reported investigations and studies. The knowledge that must be referred to examination on the seismic safety of nuclear power reactor facilities can be outlined as follows.

(1) Parameters and source mechanism of the earthquake

The earthquake parameters of the Hyogoken-Nanbu Earthquake of January 17, 1995 were publised.

The origin time: January 17, 1995 at 5:46 a.m.
Location of epicenter: 34Lo3636N, 135N,0303E
Depth of focus: 14 km
Magnitude: 7.2 (The Japan Meteorological Agency (JMA) scale)

The Hyogoken-Nanbu Earthquake has been caused by right-lateral strike-slip displacement of the fault under the east-west compression tectonics. The earthquake accompanied surface rupture (prominent right-lateral strike-slip) on the known Nojima fault, which runs along the northwest coast of the northern part of Awaji Island.

The distribution of aftershocks of the Hyogoken-Nanbu Earthquake is almost consistent with a complex system of known active faults (hereinafter called the “Rokko-Awaji fault zone”) and extending over the whole fault zone from Rokko to Awaji.

On the other hand, aftershocks of the first day after the earthquake (considered to be closely related to the earthquake source mechanism) are estimated to have been distributed in the range of about 40 km around epicenter, which is clearly shorter than the entire length, approximately 60 km, of Rokko-Awaji fault zone.

From the reasons mentioned above, the Hyogoken-Nanbu Earthquake is supposed to have been generated by the displacement of a part of Rokko-Awaji fault zone.

(2) The damages of buildings and structures

The Hyogoken-Nanbu Earthquake having the magnitude of M 7.2 is a so-called “just underneath earthquake,” which occurred at the shallow part of the earth crust, and caused severe damages.

The Hyogoken-Nanbu Earthquake caused severe damages to buildings and civil structures, including collapsed, fallen and overturned railroads and overhead bridges on the roads, as well as damaged nearly 400,000 buildings, including wooden houses, buildings of steel construction and reinforced concrete construction. Particularly some of the steel columns of steel frame buildings seem to have frittely fractured. In the coastal areas and the land-filled grounds, soil liquefaction occurred over the wide area.

The areas of seismic intensity scale VII (on the JMA scale) are distributed in the belt zone extending from Suma ward of Kobe city to Nishinomiya city. With regard to the belt-shaped distribution of damages, some reports say that the damages were caused by buried faults, but in many reports, they blamed the soil conditions in surface layers for these damages. In the rock areas around the faults, the earthquake ground motions were relatively small. In the further south soil area where damages found concentrated, the earthquake ground motions were amplified greatly. Further, it is considered that the coincidence between predominant period of earthquake ground motion and natural period of wooden houses and low and medium height reinforced concrete buildings caused the damage concentration.

With regard to the cause of the damage of buildings and civil structures, as the investigation proceeds, it was found that the horizontal seismic force rather than the vertical seismic force was predominant factor of the damages.

Many of the damaged buildings of steel frame or reinforced concrete construction were built before the enforcement of the current Building Standard Law established in 1981.

(3) Earthquake ground motion

1. Observation record of earthquake ground motion

In the Hyogoken-Nanbu Earthquake, a considerable number of strong motion of main shock were recorded in the epicentral area.

However, most of them are on the ground surface, and there is almost no strong motion record on the rock that can be directly comparable with the design basis earthquake ground motion on the free surface of the base stratum specified in Seismic Design Examination Guideline.

In the epicentral area, the strong motions that exceed 500 gal were observed, and the records over 800 gal were obtained at the sites in the central area of Kobe city, including 818 gal at the Kobe Marine Meteorological Observatory.

As for the vertical component, a peak acceleration of 556 gal was observed at the ground surface of Port Island of Kobe (landfill soil site).

2. Characteristics of earthquake ground motion

As a result of evaluation of Maximum amplitude using an empirical distance-attenuation curve to estimate the maximum amplitude of ground motion, it was indicated that the earthquake ground motions are not especially strong compared with the past great earthquakes. (Fig. 1) There were not a few observed records indicating that the peak acceleration of vertical component of ground motion was more than 0.5 times that of the horizontal component. However, many of these records were obtained on the ground surface near the seashore or of the flood plains site along river. Thus it was pointed that these sites must have received greater influence of ground surface non-linearity of surface layer subjected to the strong ground motion.

The frequency characteristics of the Hyogoken-Nanbu Earthquake can be supposed to have a relatively long period of predominant 1 second from the response spectra of earthquake ground motion observed at the Kobe Marine Meteorological Observatory.

5. EXAMINATION OF VALIDITY OF THE GUIDELINES BASED ON VARIOUS INFORMATION OF THE HYOGOKEN-NANBU EARTHQUAKE

(1) Selection of items to be examined

After studying and examining various factors of the earthquake, main conditions listed include the very strong motion of M 7.2 directly underneath large urban city located along Rokko-Awaji fault zone, observation of strong earthquake ground motion near the fault, and not a few observation data indicating the peak acceleration of vertical component was than 0.5 times as strong as horizontal component.

Based on these factors mentioned above, the items to be examined can be considered as the following three.

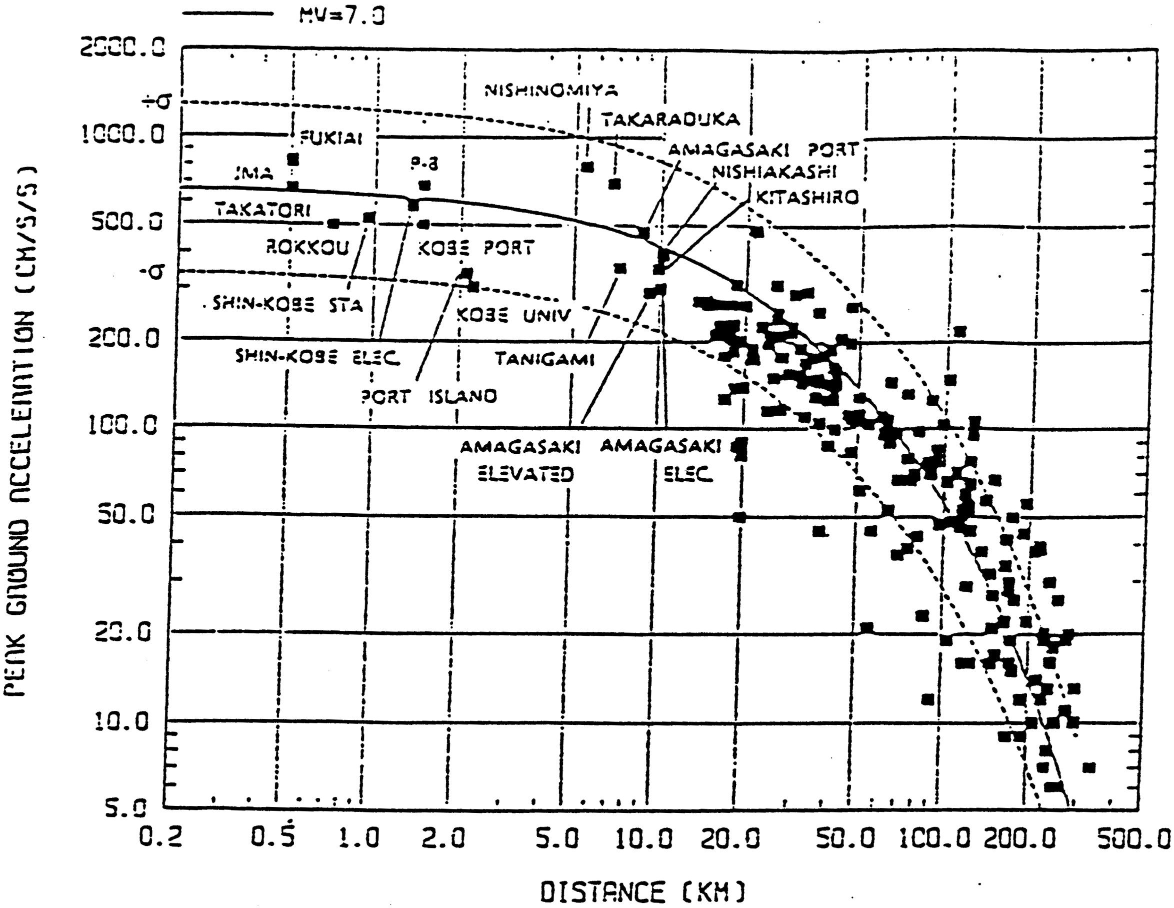

FIG. 1. Attenuation relation of maximum horizontal acceleration. The solid line indicates the attenuation relation derived by Fukushima and Tanaka (1992). The dotted line represents the range of standard deviation σ = 1. (Extracted from the outline of lectures at the meeting reporting the Hyogoken-Nanbu earthquake by the Soil Engineering Academy of Japan.

a) If there is any problem or not in the evaluation method of the design basis earthquake and the earthquake ground motion,
b) If there is any problem or not in the evaluation method of vertical seismic force, and
c) If there is any problem or not in the consideration of evaluation of active fault and magnitude of just underneath earthquake.

(2) Evaluation of earthquake and ground motion based on the Seismic Design Examination Guideline

In examining the evaluation methods of the earthquake and the earthquake ground motion, it is necessary to set up some proper site. The site conditions must be located near the epicentre, must have obtained time-history records of ground motion, and must be a site not affected significantly by the ground surface such as the liquefaction.

Taking consideration of the site conditions mentioned above, Kobe University located at Rokko-Dai-cho in Nada ward of Kobe city, which is close to the epicenter, was selected as the evaluation site.

1. Evaluation of the earthquake

Magnitude of earthquakes in the Hanshin and Awaji area were estimated based on the studies on the historical earthquakes, active faults, and the seismo-tectonic structure, followed the Seismic Design Examination Guideline. (Table I)

As the result, the earthquake that gave the greatest influence to the Kobe University site was determined to be an earthquake of M 7+3/4 assumed to have occurred in the active fault system ranging from the southeast foots of Rokko mountains to the northern part of Awaji Island from the standpoint of the seismo-tectonic structure. (Fig. 2)

Because the magnitude of the estimated earthquake is larger than the Hyogoken-Nanbu Earthquake of M 7.2, the validity of the evaluation method of earthquake based on the Seismic Design Examination Guideline is concluded to be not impaired even by referring to the Hyogoken-Nanbu Earthquake.

TABLE I. EARTHQUAKE ASSUMED AROUND KOBE AREA

Type	Assumed earthquake			Remark
	Mark	Magnitude (M)	Distance from epicenter (km)	
Histrical earthquake	E - 1	7.5	34	Earthquake in Kyoto and Kinai in 1596
	E - 2	8.4	180	Earthquake in Goki and Shichido in 1707
Active fault	F - 1	7.7	16	Earthquake caused by the active fault system extending from the south-east foots of Rokko mountains to the northern part of Awaji island
	F - 2	7.6	25	Earthquake caused by the Arima-Takatsuki tectonic line
Seismo-tectonic structure	T - 1	7 3/4	16	Intra-plate earthquake
	T - 2	8 1/2	180	Inter-plate earthquake re;ated to the Philippine Sea Plate
Just underneath earthquake	N	6.5	(X=10)	Just underneath earthquake

Note; X : Distance from hypocenter in km

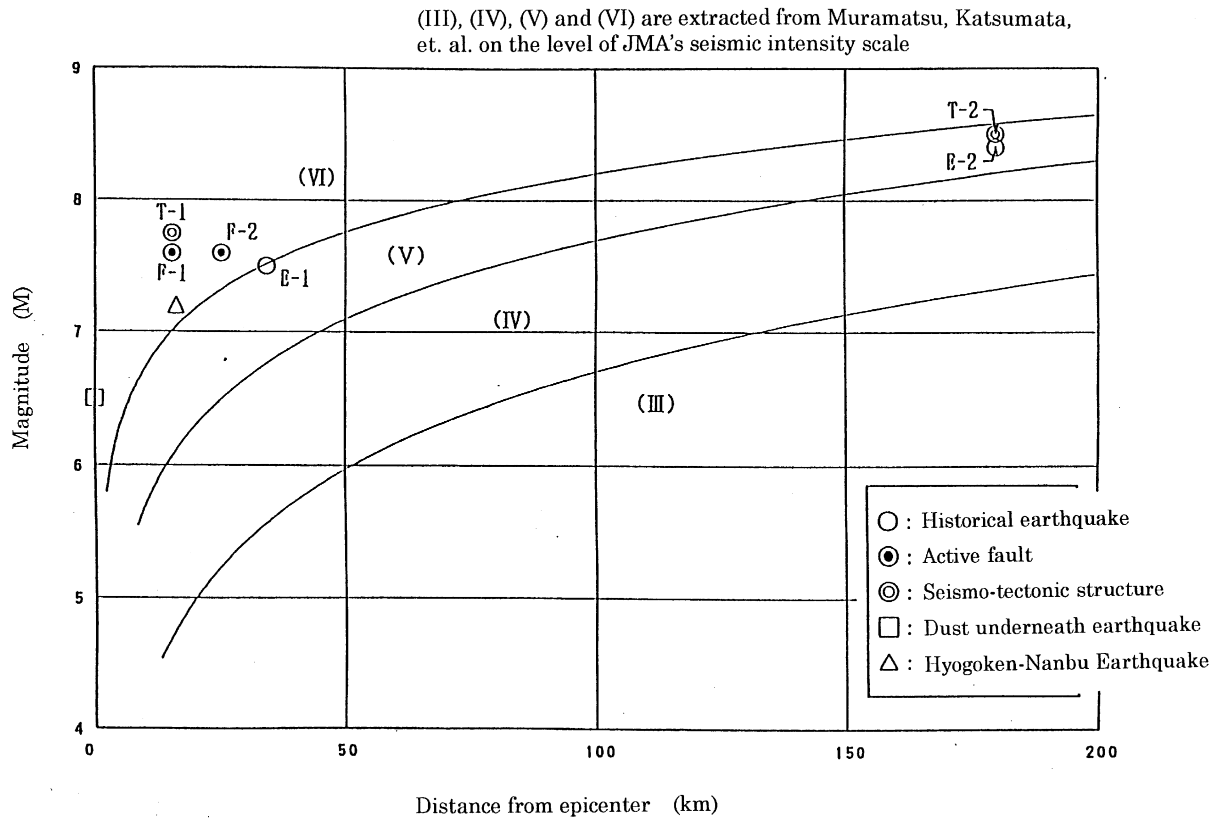

Fig. 2. Earthquakes to be considered and the effect of the 1995 Hyogoken-Nanbu earthquake. The distance from the epicentre of the Hyogoken-Nanbu earthquake indicates the distance from the centre of the areas hit by the aftershocks indicated in the "Earthquake at the Disaster — Prompt Report on Tsunami of the 1995 Hyokoken-Nanbu Earthquake" (The Meteorological Agency, January 1995).

(2) Evaluation of earthquake ground motion

Ground motions (of earthquakes estimated in Gr above) on the free surface of base stratum at Kobe University site are estimated on the standard method after OHSAKI, and the Fault model usually used when the evaluation site is close to the hypocenter. As a result of this estimation, the response spectrum of the earthquake ground motion estimated for Hanshin-Awaji area was found a larger value than that of observed ground motion at Kobe University site. (Fig. 3)

Although in the range of the long period, some value of the response spectra of ground motions obtained at Kobe University are larger than estimated ones, the validity of the evaluation method of earthquake ground motion was considered noy to be impaired for the following reasons: (a) because Kobe University site is not on hard rocks defined in the Seismic Design Examination Guideline and the influence of the amplification of subsurface layers could be possible, and (b) because nuclear facilities such as the buildings, structures, equipment, and piping sytems which are important for safety, are of rigid structure, as a principle, and their natural periods are designed in the short period range.

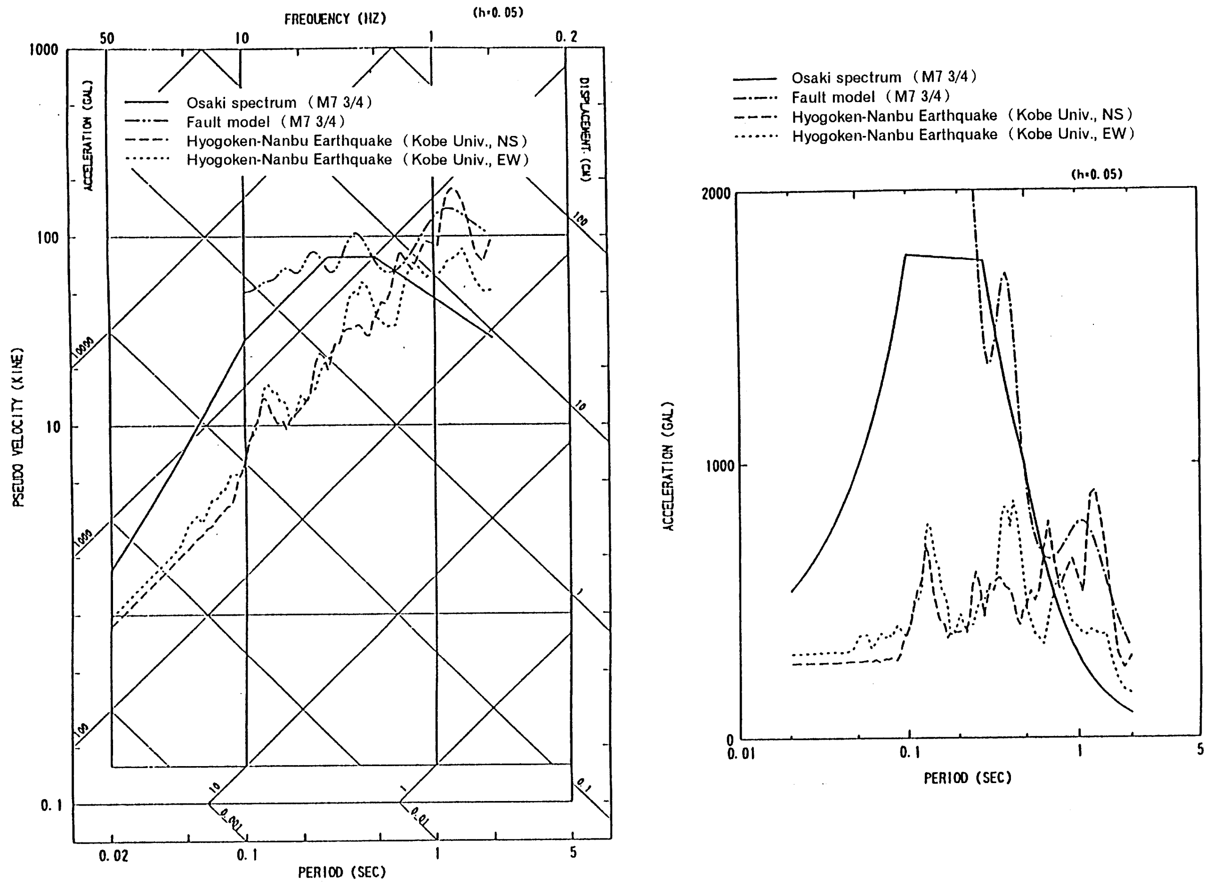

FIG. 3. Comparison of the response spectra of earthquake assumed in rock site and the earthquake ground motions recorded at Kobe University (Rokkodai-machi in Kobe city).

(3) Evaluation of vertical seismic force

1. Observed vertical ground motions at the Hyogoken-Nanbu Earthquake

An analysis was made on the ratio of the vertical and the horizontal components based on the observed records of 126 sites, excluding those at the landfill soil sites and those supposed to have strong influence of structures. The result of this analysis indicated that the ratio of the peak acceleration amplitudes of vertical and horizontal components was less than 1/2 on an average. (Fig. 4)

2. Evaluation of the vertical seismic force in the Seismic Design Examination Guideline

The Seismic design Examination Guideline requires that horizontal seismic forces shall be combined concurrently into most disadvantage mode with vertical seismic force that based on the value of 1/2 of the maximum acceleration amplitude of the basic design earthquake ground motion. In relation to above requirement, the committee investigated the ratio of the acceleration amplitudes of vertical component at the same time when the maximum acceleration of horizontal component occurred, using the data of 23 sites on which the time-history seismic waves were obtained. As a result, the average ratio was about 0.1 and the maximum value was about 0.3. They were much lower than 1/2. (Fig. 5)

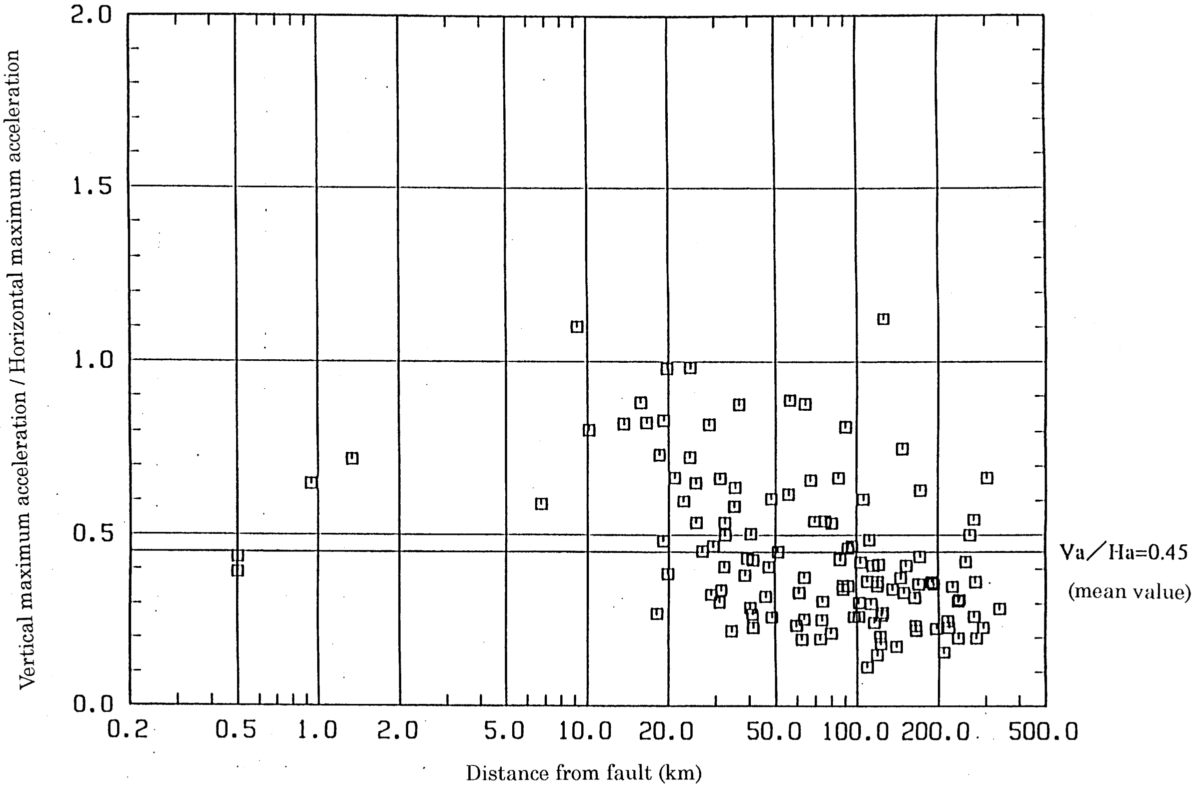

FIG. 4. Relationship of the distance from fault and the ratio between vertical and horizontal maximum acceleration.

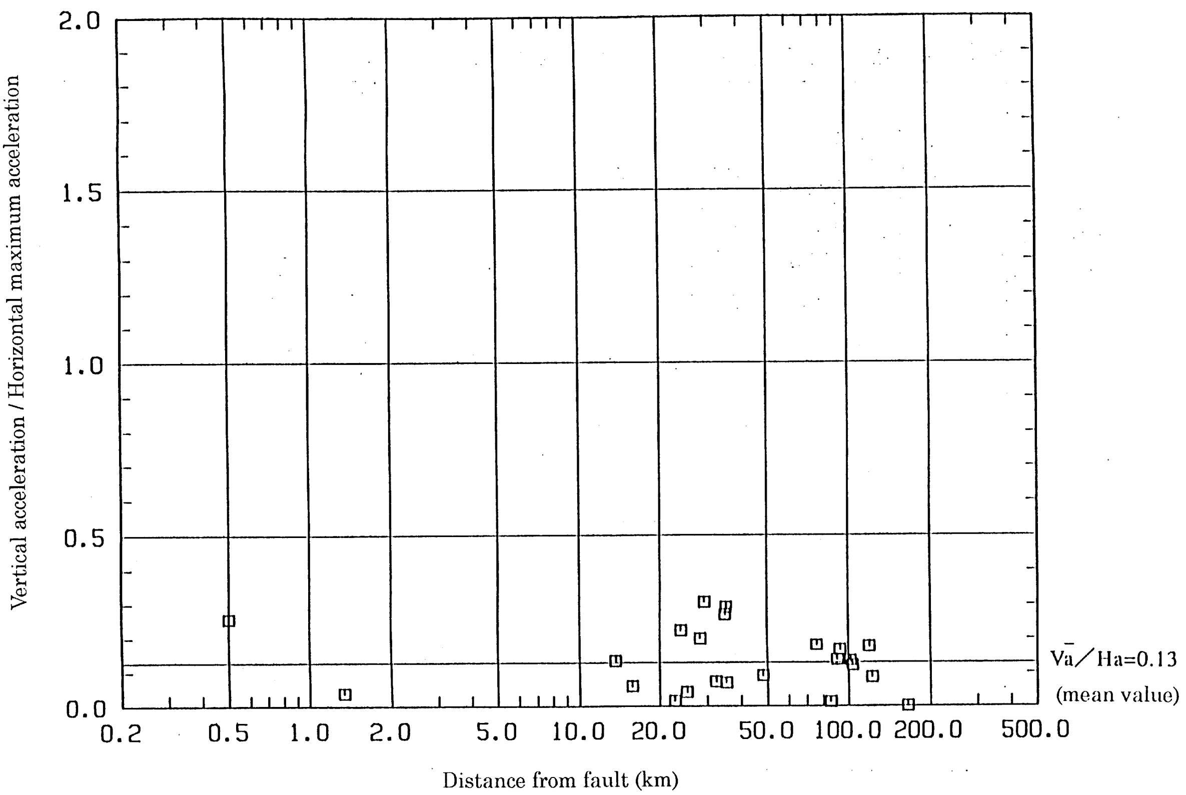

FIG. 5. Relationship of the distance from fault and the ratio between vertical acceleration when horizontal maximum acceleration occurred and horizontal maximum acceleration.

In general, it is said that vertical seismic force is not dominant to compare with horizontal seismic force, and the effect of vertical seismic force is to be small in seismic design. Therefore, most seismic design standards, including the Building Standard Law, do not specify the provisions for vertical seismic force. With regard to the damages to structures caused by the earthquake, the main factor of these damages by the Hyogoken-Nanbu Earthquake is reported to have been caused by the strong horizontal ground motion even though there might have been some influence of vertical motion.

The buildings of nuclear power reactor facilities are thick reinforced concrete shear resistant wall type structure due to the need of radiation shielding and strong horizontal seismic force for design purpose. These buildings are thus of more rigid construction compared with commercial and industrial buildings.

In addition, the vessels including the pressure vessels of nuclear power reactor facilities have high rigidity for vertical direction. The pumps are of rigid construction and the piping are appropriately supported for vertical as well as oblique directions in addition to horizontal direction so as not to easily vibrate.

For these reasons, nuclear power reactor facilities are constructed with high rigidity especially for vertical direction. Thus the effect of vertical ground motion is considered to be small for nuclear power reactor facilities.

3. Summary

From the facts mentioned above, the validity of the evaluation of vertical seismic force in the Seismic Design Examination Guideline is concluded not to be impaired even though by referring to the Hyogoken-Nanbu Earthquake.

(4) Consideration of the evaluation of active faults and magnitude of just underneath earthquake

Among the Rokk-Awaji fault zone, the earthquake return period for the active faults on Kobe side is estimated to be about 2,000 years according to the reports on these active faults. In addition, as a result of the excavation study after the earthquake, the Nojima fault has recorded another previous disturbance after twelfth century. Because the earthquake return period for these faults is shorter than 50,000 years, the validity of the concept of Seismic Design Examination Guideline, which specifies the evaluation period for active faults as 50,000 years, is concluded not be impaired by referring to the Hyogoken-Nanbu Earthquake.

Furthermore, the Seismic Design Examination Guideline requires the consideration of magnitude 6.5 (just underneath earthquake) as the design basis earthquake even when no active fault is recognized near the site. Because the Hyogoken-Nanbu Earthquake (M7.2 just underneath earthquake) occurred in a region where a complex system of active faults had been previously mapped, and the earthquake having the magnitude greater than M7.2 of Hyogoken-Nanbu Earthquake could be supposed to occur from the length of fault zone, any findings is not obtained which impairs the validity of assuming that this kind of just underneath earthquake even by referring to the Hyogoken-Nanbu Earthquake.

(5) Summary

As a result of understanding the various information of the Hyogoken-Nanbu Earthquake and by examining the matters to be discussed based on the concept of the Seismic Design Examination Guideline in detail and on the basis of knowledge obtained from the Hyogoken-Nanbu Earthquake, it is concluded that the validity of basic guidelines for securing the seismic safety of nuclear facilities of Japan is not impaired.

6. CONCLUSION

The Seismic Design Examination Committee surveyed the related guidelines on seismic design, selected the items to be examined, and examined on those items based on the knowledge obtained from the Hyogoken-Nanbu Earthquake. As a result, the Committee confirmed that the validity of the guidelines regulating the seismic design of nuclear facilities is not impaired even though the basis of the Hyogoken-Nanbu Earthquake.

However, the people related to the nuclear facilities may not be content with the above result, but continuously put efforts in doing the following matters to improve furthermore the reliability of seismic design of nuclear facilities by always reflecting the latest knowledge on the seismic design.

1) The people related to nuclear facilities must seriously accept the fact that valuable knowledge could be obtained from the Hyogoken-Nanbu Earthquake, try to study and analyze the obtained data, and reflect the results of investigations, studies, and examinations conducted appropriately to the seismic design of nuclear facilities referring to the investigations and studies of related academic societies.
2) Research and test investigations are to be performed to further enhance the seismic design.
3) Proving demonstration of seismic resistance for nuclear facilities are needed all the more.
4) The information on seismic design must be provided to public in general and efforts must be put to the international exchange as well as to the joint study of international basis.

It goes without saying that it is important for the nuclear facilities to obtain public acceptance on the safety of facilities as well as to secure seismic design and sufficient safety standard. The people related to nuclear facilities are requested to put incessant efforts to the seismic safety of nuclear facilities and to develop much more national reliance.

THE K-NET — A YEAR AFTER

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Abstract

We started to release the K-NET strong-motion data from June 1996 and about one year passed. In this article, we report the development of K-NET and some applications using the K-NET information released on the Internet.

Key Words: K-NET, Strong-motion

1. DEVELOPMENT OF K-NET

1.1. K-NET information on the Internet

We offered the K-NET information through graphical user interface on the World-Wide-Web in 1996. From April 1997, the text pages of K-NET are available on the Internet for more quick distribution of the K-NET data.

1.1.1. Strong-motion records

We released the strong-notion records obtained by the K-NET on the Internet. However, the Internet capacity in this country is so poor that it becomes impossible to get the K-NET data on the Internet just after moderate earthquake occurred. For example, in case of the Izu-Hanto-Toho-Oki earthquake swam occurred in March 1997, a part of users did not connect to the K-NET Internet site. To settle this problem, we constructed two mirror sites of our control center and started to release the K-NET information from these mirror sites from April 1997. Now, users can get the K-NET data from the following three Internet sites.

1) Control Center http://k-net.bosai.go.jp
2) Mirror site #1 http://k-net.ostec.co.jp
3) Mirror site #2 http://k-net.geophys.tohoku.ac.jp

The mirror sites #1 and #2 were installed in Osaka and Sendai cities, respectively. Also, from April 1997, users can download all strong-motion records obtained from a specific event in a lump sum. Such a data retrieve is possible for events selected by users.

1.1.2. Site information

We set up K-NET ftp site at October 1996. From the ftp site, users can get all the soil data of K-NET stations in a lump sum. Also, the two mirror sites set up the ftp sites and started the service of K-NET data release from April 1997. The address of these ftp sites is obtained by changing the header part of Internet address from http to ftp. The ftp sites are also possible to release the K-NET strong-motion data. In this case, the data set is constructed with data directories, which correspond to earthquake origin times.

1.1.3. Maximum acceleration map

Usually, we make a maximum acceleration map when we get more than 50 three-component seismograms for an earthquake. Figure 1 shows a sample. This map was obtained from the Hiuganada earthquake of October 19, 1996. The JMA magnitude is 6.6. The contour lines of acceleration are calculated by using Splines interpolation. From these maps, we can interpret the local characteristics of maximum acceleration.

In 1996, we released a viewer program, which plots the K-NET seismograms. In 1997, we are going to revise the viewer program and release the following application programs:

1) program for the calculation of velocity and displacement seismograms,
2) program for the calculation of Fourier and Power spectra,
3) program for the calculation of response spectra, and
4) program for the calculation of JMA seismic intensity defined by JMA in 1996.

These programs can plot the calculation results.

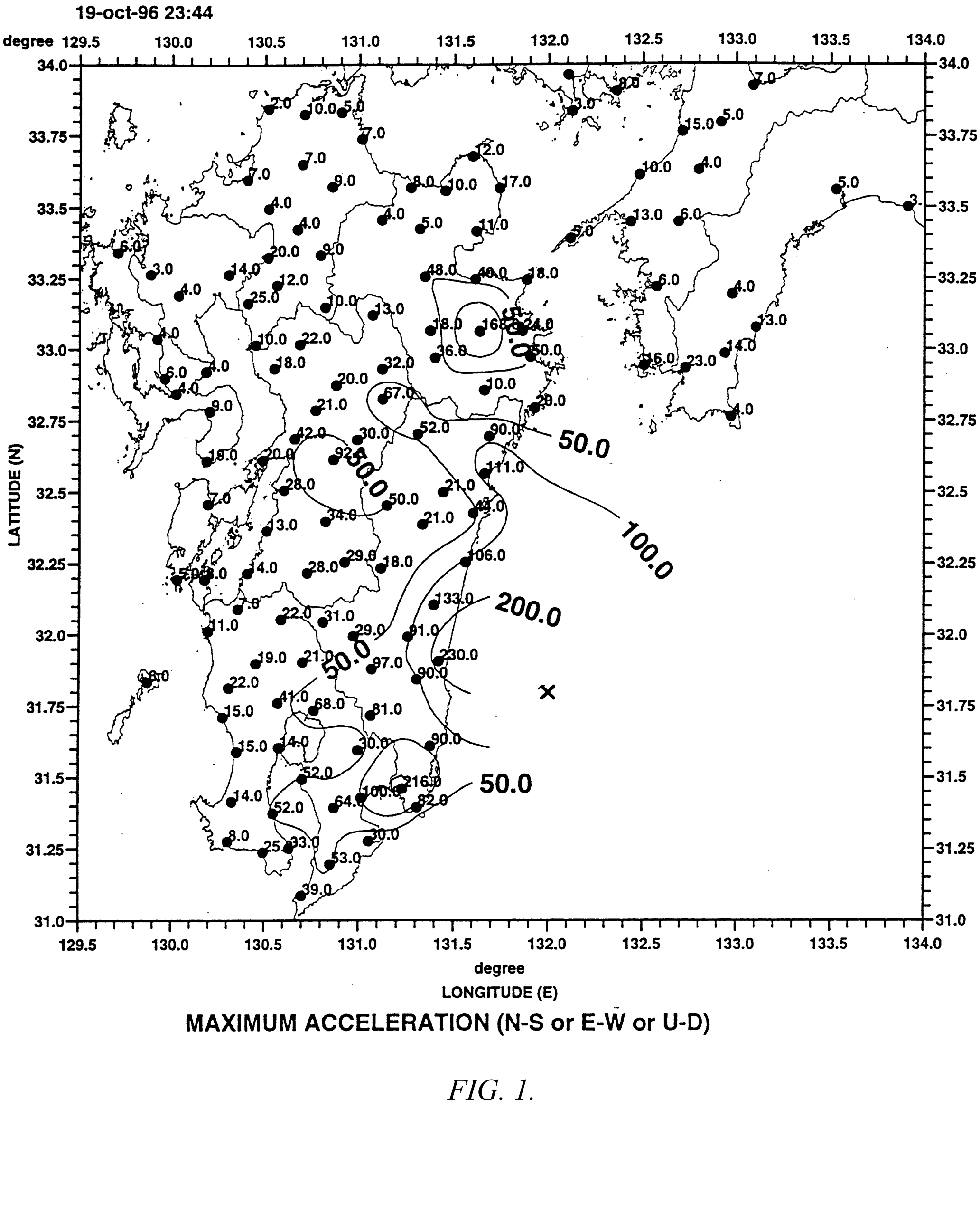

FIG. 1.

1.1.4. Application software

1.2. Off-line release of the K-NET information

From April 1997, user can copy the K-NET strong-motion data to user's MO and/or DAT at the control center and Mirror site #2. This is a self-service. Also, we distributed the strong-motion data obtained in 1996 by CD-ROM. Such a service is going to be done every half year. The recording characteristics of acceleration seismograph, which used in the K-NET, were also published.

1.3. Use of K-NET

The K-NET started to release the strong-motion data on the Internet from June 3, 1996. Figure 2 shows the number of accessed pages for every day. Usually, the release of K-NET data on the Internet is made within 24 hours after an earthquake occurred. For Example, in case of the earthquake of March 16, 1997 (M = 5.6), we obtained 209 three-component seismograms and released these data within about 3 hours. In such a case, the number of accessed pages became from 1 000 to 3 000.

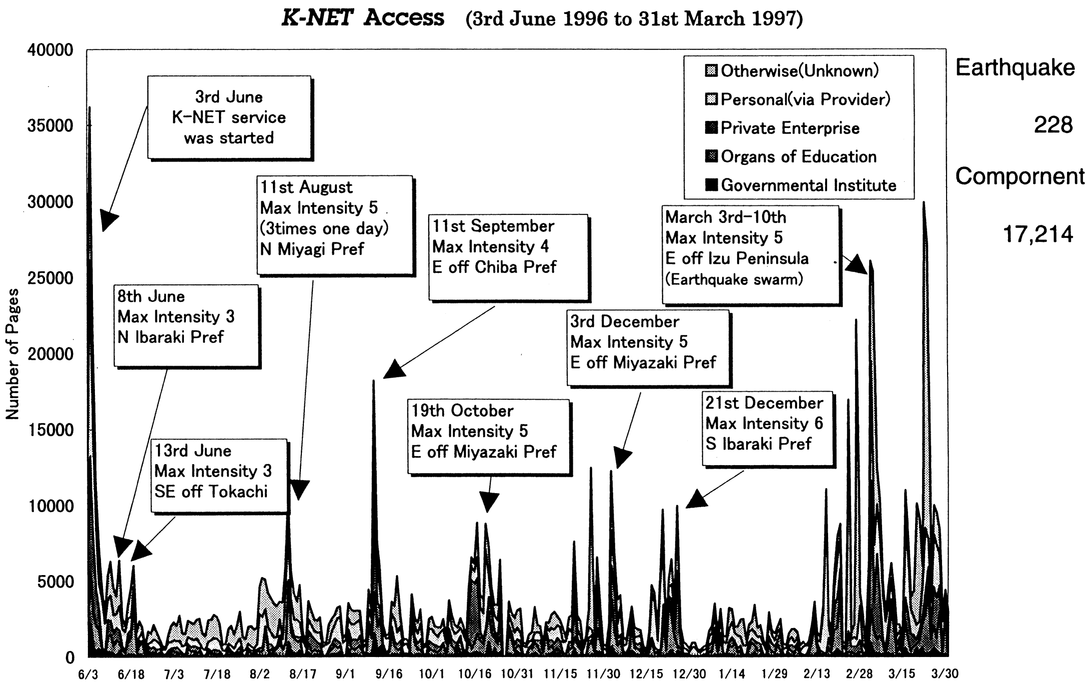

FIG. 2.

2. APPLICATION

Since the K-NET starts, about one year passed. Some application studies using the K-NET data were reported during this period. Representative applications are as follows.

2.1. Monitor for strong quake

The K-NET 95, an acceleration seismograph used in the K-NET, has two communication ports. One is directly connected to a modem belong to a local municipal government. Among them, about 600 seismographs were incorporated in the Seismic Intensity Information Network of the Fire Defense Agency (FDA). This network consists of about 3 400 stations. Just after an earthquake, the FDA may get the seismic intensity information in the hypocentral area within several minutes. The data will help local governments to assess the extent of the affected area and to manage their disaster response efficiently. This network is going to start in 1997.

The students of Tokyo University are opening the seismic intensity information on the Internet by using the K-NET data. The address is http://yagamo.u-tokyo.ac.jp/426/shindo.html. They show seismic intensity map just after the K-NET data are released. They used a formula for calculation of JMA seismic intensity defined by the JMA in 1996.

2.2. Utilization as a buck up network

Many research institutes are performing small-scale array observation on the basis of their original purposes. The K-NET supports these observations as a back-borne network. Namely, the K-NET supports the data, which can not be covered in their arrays. For example, the K-NET data obtained from the earthquake of December 21, 1996 (M=5.4) helped an array observation deployed in Tama area, Tokyo, to interpret the existence of remarkable S-waves totally reflected in the Philippine Sea Plate.

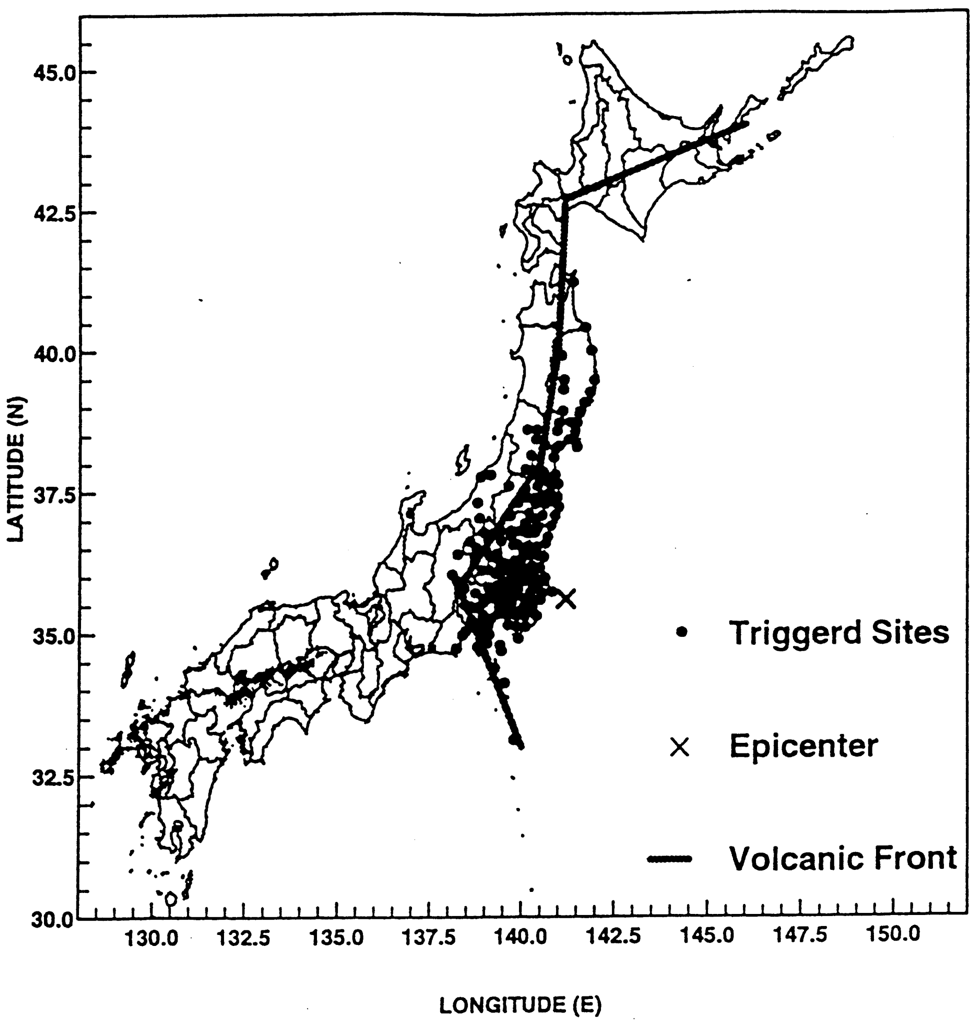

FIG. 3.

2.3. Studies on seismic wave propagation

The K-NET is suitable on the seismic wave propagation in a hypocentral area with a radius of about several hundred kilometers. Study on the regional attenuation of seismic wave is one of the reasonable research subjects. For example, the K-NET data obtained from earthquake of September 11, 1996 (M=6.2) showed that S-waves severely attenuate when they pass through the volcanic front in Kanto area as shown in Figure 3. Solid circles in this figure show the triggered stations during the earthquake.

2.4. Studies on near field seismograms

As the K-NET seismographs are distributed all over Japan with a station distance of about 25 kilometers, they have the opportunity which obtains near field seismograms. In 1996, the K-NET95 installed at the Naruko station recorded the main and after shocks during the Miyagi-ken Hokubu earthquake. The JMA magnitude of main shock is 5.9. The largest acceleration is over 700 Gals at the site. Some research institutes tried to get the after shock records around Naruko site and tried to interpret the main shock seismogram obtained at the Naruko station.

PLAN FOR 3-D FULL-SCALE EARTHQUAKE TESTING FACILITY

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Abstract

Based on the lessons learnt from the Great Hanshin-Awaji Earthquake, National Research Institute for Earth Science and Disaster Prevention plan to construct the 3-D Full-Scale Earthquake Testing Facility. This will be the world's largest and strongest shaking table facility. This paper describes the outline of the project for this facility. This facility will complete at early 2005.

1. BACKGROUND

The Great Hanshin-Awaji Earthquake (January 17, 1995) gave us serious damages. Many people lost their life or injured by the earthquake. Many buildings and other structures were destroyed.

Especially modern society, which consists of the advanced, complex facilities and concentrated population and functions, has proved to be weak against natural disaster.

To contribute immensely to reducing earthquake disaster, it is indispensable to perform seismic tests on real-size objects and large-scale model of structures. Through, these tests we can elucidate the process of structure destruction and minimize its damage.

On this purpose, we plan to construct the 3-D Full-scale Earthquake Testing Facility, which can be expected to promote the international cooperation.

2. OUTLINE OF THE FACILITY

(1) Specification of facility

The facility (3-D Full-scale Earthquake Testing Facility) can simulate the earthquake grand motion as same size as the Great Hanshin-Awaji Earthquake.

- Main specifications

Dimensions of shaking table			20m × 15m
Maximum model weight			1 200 ton
Maximum displacement	horizontal	X:	± 100 cm
		Y:	± 50 cm
	vertical		± 50 cm
Maximum velocity	horizontal	X:	200 cm/s
		Y:	100 cm/s
	vertical		70 cm/s
Maximum acceleration	horizontal	X,Y:	0.9G
	vertical		1.5G

(2) Planed construction period: 1998 ~ 2004 (Japanese fiscal year base)

(3) Major research subjects

Liquefaction of sand layer
Reinforced concrete building
Bridge
Liquid tank
Behavior of structure while liquefaction
Interaction between non-liquefied ground and structure

(4) Usage

The facility will be available for use to domestic and international research organizations and researchers.

3. DEVELOPMENT OF TECHNIQUE FOR VIBRATION SYSTEM

Prior to construction of the 3-D Full-scale Earthquake Testing Facility, four years project for horizontal and vertical vibration system and the prototype of the facility has been developed since 1995.

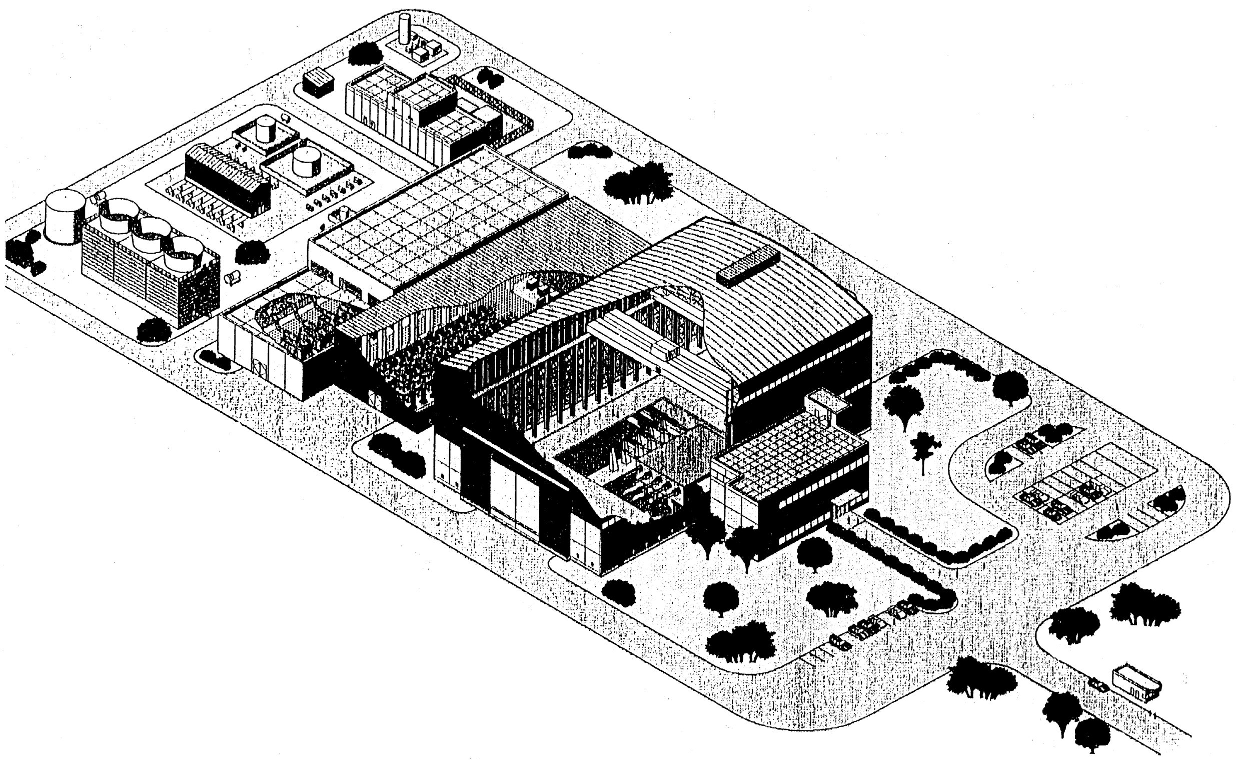

FIG. 1.

- Main Specifications -

Item	Horizontal X	Horizontal Y	Vertical Z
Table Size	**20 m x 15 m**		
Driving Method	**Accumlator Charge / Electro-Hydraulic Servo Control**		
Maximum Test Weight	**1200 tonf**		
Maximum Acceleration (at Max. weight)	**0.9 G**	**0.9 G**	**1.5 G**
Maximum Velocity	**200 cm/s**	**100 cm/s**	**70 cm/s**
Maximum Displacement	**±100 cm**	**±50 cm**	**±50 cm**
Maximum Overturning Moment	**≧15,000 tonf · m (at Az=1G)**	**≧15,000 tonf · m (at Az=1G)**	——

FIG. 2. 3-D full scale earthquake testing facility — main specifications.

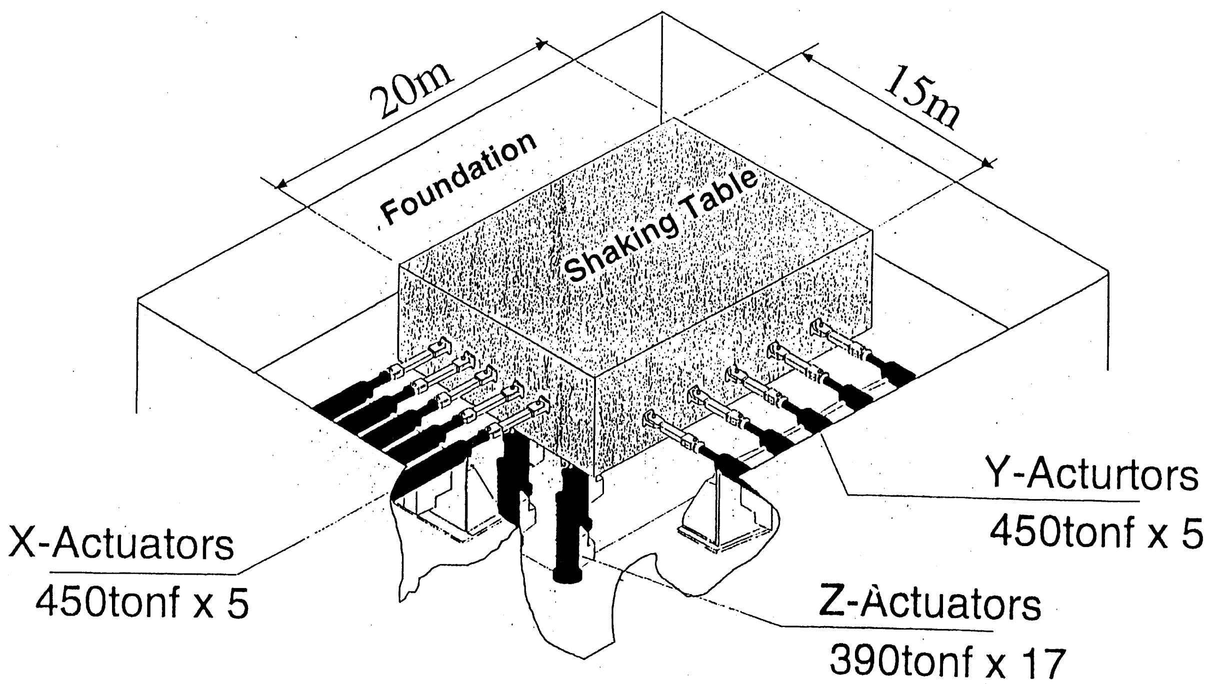

FIG. 3. 3-D full scale earthquake testing facility — outline of main part.

SEISMIC HAZARD ASSESSMENT IN INTRA-PLATE AREAS AND BACKFITTING

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Abstract

Typically, fuel cycle facilities have been constructed over a 40 year time period incorporating various ages of seismic design provisions ranging from no specific seismic requirements to the life safety provisions normally incorporated in national building codes through to the latest seismic nuclear codes that provide not only for structural robustness but also include operational requirements for continued operation of essential safety functions. The task is to ensure uniform seismic risk in all facilities. Since the majority of the fuel cycle infrastructure has been built the emphasis is on re-evaluation and backfitting. The wide range of facilities included in the fuel cycle and the vastly varying hazard to safety, health and the environment suggest a performance based approach. This paper presents such an approach, placed in an intra-plate setting of a Stable Continental Region (SCR) typical to that found in Eastern Canada.

1. INTRODUCTION

The understanding of seismic hazard changes with time. Recent earthquakes, geological findings, combine to increase the assessment of seismic risk in nuclear facilities. New nuclear power plants designed and built in the 1970s and early 1980s required ever increasing levels of seismic protection. As reactor seismic engineering increased, concern was also given to non-reactor nuclear facilities. Fuel cycle infra-structure be it tailings dams, fuel conversion plant, irradiated fuel storage, heavy water plants or research facilities came under increasing pressure to be seismically qualified to today's standards. Much of the nuclear fuel cycle facilities is older. Most were designed to the building codes of the day. Building codes provided a level of structural protection but left out an important component of reactor seismic philosophy. That philosophy requires that in addition to structural and occupant safety the facility can continue to meet its safety functions during and after an earthquake with sufficient staff in place to continue the needed monitoring and control of the safety systems as well as provide other needed emergency services.

2. TYPE OF FACILITIES

Nuclear fuel cycle facilities include:

- Mines and mills
- Fuel conversion plants
- Fuel fabrication
- Irradiated fuel (pool storage/concrete canisters)
- Isotope separation facilities
- Radioisotope laboratories
- Manufacturing facilities using prescribed substances
- Research facilities
- Heavy water plants

The complexity of engineered systems varies from passive retaining structures, to fuel storage, to research facilities and finally to complex chemical plant that include not only radiological but a mix of harmful chemical hazards. The radiological hazard and the chemical hazard (e.g. H_2S or $S0_2$ in case of heavy water plants) can cause both on-site and off-site harm. For earthquakes damage can be caused from failure of internal equipment and from induced failures brought about by outside hazards such as upstream dam failures, landslides, tsunami. Failure of outside services such as water suppliers and electrical grid can also increase probability of facility damage or impede site staff and external resources from carry out mitigating actions and rescues.

3. HAZARD TO BE CONTROLLED

In severe earthquakes it must be assumed that normal processes will cease to function or become intermittent. The facility will need to meet its safety functions. For reactors, the safety functions are to shutdown, provide continued core cooling, isolate and contain potential radioactive sources and provide sufficient monitoring and control activities to stabilize and eventually return to service the reactor. Fuel cycle facilities vary in complexity and it is more difficult to make general rules but if one takes an isotope separation facility as an example, it is clear that an orderly shutdown of normal operation should be provided for, that containment or appropriate venting be required, that sufficient monitoring be available to allow plant damage control assessment and the formulation and execution of appropriate mitigating actions. The inplant hazard that needs to be controlled has to be specified ahead of time and the defences specified and provided for.

Experience in examining damage and non-damage of industrial facilities in actual earthquakes has shown that earthquake induced accidents may have subtle origins and may have little to do with the normal inertia induced failures that one associates with structures or components. The meaning here is best illustrated in the following example (Reference 1).

The more likely causes of an earthquake-induced accident (after the basics are covered, such as ensuring adequate anchorage of equipment) are more subtle effects such as earthquake induced sloshing of the lubricating oil in the oil pan of an emergency diesel generator (much like the water in a glass sloshes when we gently slide the glass back and forth on a table). When the surface of the sloshing oil is below its normal level, the oil level sensor in the oil pan of the diesel engine (falsely) senses that the oil level is too low, and aborts any in progress automatic start of the diesel engine to prevent damage to its bearings because of a (perceived but not real) lack of lubricating oil. This may require an operator to go to the diesel and determine what went wrong, and manually restart the diesels. The safety question is whether operators can recover quickly enough.

Note that the failure to start is not caused by lack of seismic qualification of any component, or by any earthquake damage. Note also that this effect is not fixed by strengthening the components so they have a higher seismic capacity. It is a weakness in the oil level sensing logic to apparent, but not real, loss of lubricating oil. One fix is to install logic that can recognize the difference between oil sloshing and loss of oil.

This example also illustrates that the more likely plant weaknesses are probably in the places we are not looking. Subtle effects, such as in this example, are found by qualified engineers who are familiar with the effects of past earthquakes on power and industrial facilities, working with plant personnel familiar with the intimate details of how nuclear plant equipment actually works.

4. TYPICAL STRUCTURES, SYSTEMS AND COMPONENTS

Nuclear facilities typically have massive reinforced concrete, low height, structures that can be screened out as robust in a seismic evaluation. Structural steel framing, overhead cranes in particular will need a closer look. Concrete block or brick walls by themselves or acting as supports for instruments or the routing of services whether attached or close to block walls needs to be investigated. Electrical distribution systems and associated control and protection gear, batteries, relays, motor control centres will need careful review. What characterizes all distribution systems whether they are air, gas, liquid, or ventilation systems, is that the “weak-link” effect requires, that for effective seismic protection these systems must either be qualified in total or qualified isolation must be provided.

Material handling and liquid storage is a significant part of many fuel cycle facilities. Conveyors, storage racks, tanks for bulk and liquid storage of all sizes abound that will require a systematic review.

Review of older plants usually show sign of significant hazards in diverse place due to potential rolling stock, gas bottle storage, fuel storage for day tanks of emergency diesels, etc. Seismic operational procedures need to be established and a commitment to implement such procedures to minimize these types of hazards.

5. EXISTING METHODS OF SEISMIC PROTECTION

Whether inherent or explicit a level of seismic protection exists in all facilities. Building codes have provided horizontal force resistance based on inertial mass. Later additions of systems (e.g. piping loops, I&C improvements) have been done to modern codes including test qualified components. Plant engineers have become more sensitised to provide seismic protection and usually chose seismically capable equipment even if formal qualification spectra are not included in the purchasing specifications. Much of the equipment in fuel conversion facilities operates in a rugged factory setting with background vibration environment due to rotating equipment, material handling equipment and cranes that require shock tolerant components that should do well in future seismic ground motion. Diesel and other prime movers for emergency power generation usually come from a marine application environment and may be assumed to be vibration resistant. “similar application” argument are recognized in some seismic qualification standards such as, for example, the N289.4 Canadian Standards Association Code for Seismic Testing.

6. UNIQUE AND CRITICAL AREAS FOR SEISMIC PROTECTION

It is not normally necessary for all parts of a facility to be seismically qualified. Based on a consideration of health, safety and protection of the environment a subset of functions are chosen to be qualified (for existing plants) or designed in new plants or new additions to existing plants. Typical safety functions include process shutdown, containment or isolation, ventilation control and continuation of monitoring and emergency control. For some processes continued cooling (or heating) may also be required. The key point is that a subset of the normal systems must be intact and functioning to meet the seismic requirements. These requirements usually also generate additional requirements for the building enclosure and other systems to prevent collapse or large distortion, to prevent collateral damage and to minimize threat of fire or internal flooding.

7. EXPERIENCES FROM PAST EARTHQUAKES

While no significant ground shaking has occurred at the location of nuclear facilities in Canada, several of the significant earthquakes have taken place in the Stable Continental Region of Eastern Canada. In 1944, at Cornwall Massina, a 5.7M earthquake occurred under two highly industrial cities that were in full war time production. Damage to individual homes were considerable but the factories (aluminium production, chemicals) were essentially unharmed. In Miramichi, New Brunswick, a similar sized earthquake in 1982 caused minor cessation of mining operations and set off alarm circuits in electric power plants and process factories. The larger Saguenay earthquake of 1989, 6M, occurred close to large aluminium industrial towns, an Air Force Fighter Base and caused potentially severe disruption due to power outages. Some toxic gases were released due to malfunctioning valves and electrical switchgear damage occurred. At the Air Base no real damage occurred but simultaneously electrical grid failure, uncommanded activation of fire alarms did cause moments of uncertainty. The loss of the power grid had the potential to cause irreversible damage to aluminium melt as electric heating was no longer available. Fortunately power was restored within 4 hours.

These examples are given to illustrate that seismic qualification is necessary but that it need not be a major retrofit program. What is needed is a careful review of what is important for seismic safety and then a systematic program to review plant status and operations to find areas of inadequacy and correct these.

The preceding discussion sets the background for creating a unified regulatory requirement for new and existing facilities. The underlying philosophy is that seismic protection is scaled to the risk that the facility can generate. That risk is site dependent, for defining future seismic ground motion, and facility dependent. Figure 1 outlines what is required and this is a systematic dispositioning process for seismic weak-links. Figure 2 encapsulates the regulatory framework.

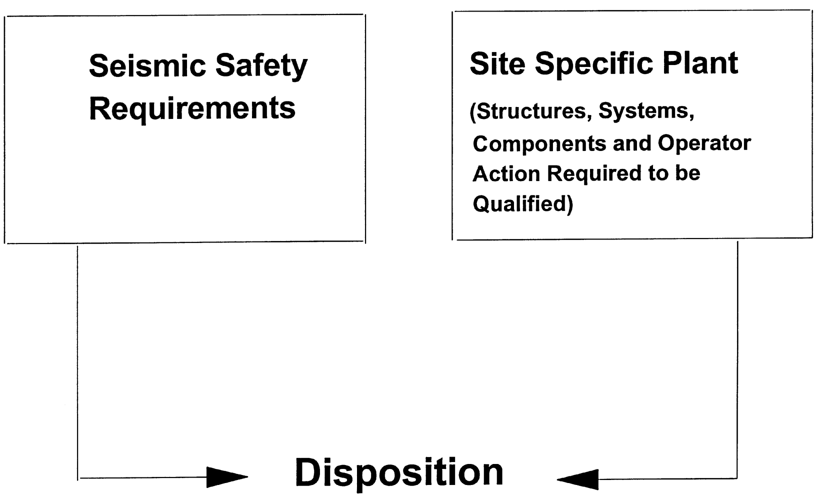

FIG. 1. Assuring seismic accuracy.

What Needs to be Achieved	Assurance that facility poses no undue risks for earthquakes that are likely to happen within the region containing that facility
Safety Requirements	shutdown contain continuation of needed system services power - cooling - ventilation monitoring and control
Seismic Hazard Determination	level of ground shaking faulting consequential damages lessons learned from earthquakes in industrial area (e.g. Kobe) - accident scenarios
New Designs	protection level - probability of accedence of site ground motion - consequences of accident scenarios - confidence in protection features analysis - level of sophistication testing - shake table - like minded applications commissioning - area checks
Existing Designs	Review Level Earthquake Success Path Qualification Techniques - experience basis - similarity testing - analysis Retrofitting
Post Earthquake Activities	Operation during and after Operator Requirements Rescue Consequential Damage Control (e.g. fire) Psychological concerns After shocks

FIG. 2. Providing a uniform set of requirements from the regulatory perspective.

8. A UNIFIED REGULATORY REQUIREMENT FOR NEW AND EXISTING FACILITIES

It is important that a comprehensive seismic protection approach which gives requirements first, then assigns barriers. The nuclear facility must continue to meet functional safety requirements both during and after the seismic event. The functional safety requirements must be clearly outlined, that is, it must be stated what needs to be protected and why.

A Success Path must be described which satisfies the Seismic Safety Requirements. The Success Path defines these structures, systems, equipment and human actions that are necessary to meet the Seismic Safety Requirements during and following the earthquake. All other systems may be assumed to have ceased providing their intended design functions.

The Success Path includes sufficient equipment and operator actions to contain sources of radioactivity and other hazards that are required to be controlled and prevent significant release. The selection of the success path is a joint responsibility of the operators, system engineers and seismic qualification engineers. A single success path is ultimately all that is required. It may be prudent to assign several paths for redundancy.

8.1. Seismic safety requirements

Typical statements in the Seismic Safety Requirements are:

- Maintain confinement of radionuclides
- Maintain monitoring capability so that leakage continues to be within acceptable values and if not, to be able to control it.
- Return to operational state those systems needed to minimize collateral damage caused by the earthquake (e.g. fire).
- Return to operational state those systems needed to control radioactive releases following the DBE.

8.2. Seismic success path

A typical Success Path ensures that the safety requirements are met. The Success Path includes structures, components (both active and passive) and operator actions.

From the Success Path the seismic classification list giving the structure and equipment to be qualified and the qualification level (e.g. DBE A, DBE B) are derived.

8.3. Protection level (assigning a recurrence internal)

The level of seismic protection required will need to be set based upon the need to ensure that the seismic safety requirements will be met. For example, for operating nuclear power plants in Eastern Canada a site-specific uniform hazard spectrum at a recommended median of 10^{-4} per annum has been suggested (Ref. 2).

9. CONCLUSIONS

Maintaining acceptable seismic risk for nuclear facilities requires a continuous effort. The understanding of seismic hazard changes with time; the plant design changes with time. Since the majority of the fuel cycle infrastructure has been built the emphasis is on re-evaluation and backfitting. The wide range of facilities included in the fuel cycle and the vastly varying hazard to safety, health and the environment suggest a performance based approach. This paper presents such an approach, placed in an intra-plate setting of a Stable Continental Region (SCR) typical to that found in Eastern Canada.

REFERENCES

[1] The use of Seismic PRAs in Safety Evaluation of Nuclear Power Plants, Prepared for: Canada Department of Justice, Toronto Regional Office, *Prepared by: Paul Smith, The Readiness Operation, August 7, 1993*

[2] INFO-0419: Pickering Seismic Safety Margin - Methodology, Dr. A. Ghobarah, *Dr. A.C. Heidebrecht, Dr. W.K. Tso,* June 1992

[3] Seismic Hazard in Southern Ontario, Geomatrix Consultants, August 1997, A systematic review of the geology, seismicity and tectonic setting surrounding the Pickering and Darlington nuclear stations, the potential for generating future earthquakes and the effect of these earthquakes on site ground motion.

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DEVELOPMENT OF A GRADED APPROACH TO NATURAL PHENOMENA HAZARD DESIGN AND EVALUATION OF RADIOACTIVE WASTE AND SPENT FUEL STORED AT NUCLEAR POWER PLANTS

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Abstract

Nuclear safety related structures, systems and components, SSC, at large commercial nuclear power plants other than those applicable to reactor safety have in general not received the attention and detailed loading and behavior criteria use for reactor design safety. Such systems include spent fuel storage and radioactive waste storage and processing. In this paper is a suggested grading of design bases for natural hazards to be applied to such facilities commensurate with their radioactive risk. They are applicable to the full range of safety related SSC which are determined by the inventory of radioactive isotopes and the unmitigated doses at appropriate plant and site boundaries.

1. INTRODUCTION

A great deal of effort over the past 35 years has been expended to develop design and review procedures associated with the seismic adequacy of the reactor coolant system and other reactor safety systems of nuclear power plants worldwide. However, there are other systems associated with gaseous, liquid and solid radioactive waste processing and storage and spent fuel storage at NPP sites for which the seismic evaluation criteria developed for the reactor system in general because of its very high hazard level may not be appropriate. In addition, there are other natural phenomena hazards such as extreme wind and flooding hazards which have not received commensurate attention. This paper is an attempt to address design and evaluation procedures for nuclear systems at NPP sites other than the reactor for seismic and all the nuclear safety related systems for other natural hazards such as wind and flooding.

2. THE GRADED APPROACH TO SEISMIC DESIGN

In addition to seismic requirements defined for the reactor systems at nuclear power plants it has long been the practice to define somewhat lesser seismic requirements for waste storage and processing Structures, Systems and Components, SSC at the plant. These different requirements are defined in the USA in Regulatory Guide 1.143 which is attached hereto as The attachment. The SSC associated with these waste systems are either designed for normal National Building Code requirements or an Operational Basis Earthquake (S_1) or a Safe Shutdown Earthquake(S_2) taken equal to one half that designated for the reactor systems. The allowable stresses for such waste related SSC under seismic loading typically have a 1.33 increase in the allowable stresses from the normal allowables, while the allowable increase for reactor SSC is typically taken as 1.0 factor for the S_1 earthquake and a 1.6 factor for the S_2 earthquake.

For the spent fuel storage facilities at nuclear power plants, the same seismic design and evaluation criteria used for the reactor is typically used for the spent fuel storage. However, there are several spent fuel storage facilities in the US operated by the US Department of

Energy where somewhat less seismic requirements have been used typically associated with lesser earthquakes which have a probability of occurrence at least one order of magnitude higher than that specified for a large power reactor coolant and safety systems.

As a result of the consideration of natural hazard design of nuclear facilities other than large power reactors, a graded approach has been developed for the design of those facilities by the US Department of Energy, (US DOE). In Section 3 and Table 1 can be found a summary of the author's interpretation and integration of the US NRC[(1,2)] and US DOE [(3,4)] criteria for use in evaluation of existing and new nuclear facility SSC. It should be noted that Table 1 also addresses extreme wind and flooding since in some instances these natural hazard loading phenomena could control design adequacy of safety related structures and in some instances systems and components particularly in relatively low intensity seismic sites located in Northern Europe and Eastern South America.

3. SUGGESTED DESIGN CATEGORIZATION OF SAFETY RELATED STRUCTURES, SYSTEMS AND COMPONENTS IN NUCLEAR FACILITIES

3.1. Design Category 4S

A SSC shall be placed in a Design Category 4S if it is necessary to prevent or mitigate the consequences from design basis hazards to a facility which has a radiological inventory in excess of Threshold II quantities given in Table II[(5,7,8)] where the unmitigated consequences are associated with:

- unmitigated[1] off-site dose in excess of 25[2] REM
- the facility system or process being evaluated is high energy having the potential for rapid and widespread dispersion of highly toxic radiological, chemical or biological material whose release could have serious potentially life threatening health or environmental effects at the site boundary.

3.2. Design Category 4S

A SSC shall be placed in a Design Category 4 if it is necessary to prevent or mitigate the consequences of design basis hazards to a facility which has a radiological inventory in excess of Threshold II quantities given in Table II[(5,7,8)] where the unmitigated consequences are associated with:

- not in Design Category 4S
- unmitigated[2] off-site dose in excess of 25[3] Rem
- unmitigated[2] on-site dose in excess of mandated limits for co-located workers
- the facility stores or processes radiological, chemical or biological material or waste whose unmitigated air borne release could have serious potentially life threatening health or environmental effects at the site boundary.

3.3. Design Category 3

A SSC shall be placed in Design Category 3 if it is necessary to prevent or mitigate the consequences of design basis hazards to a component or system which has a radiological

[1] In some instances SSC are so robustly designed or constructed or have sufficient redundancy and diversity that their mitigation capabilities are assumed in the dose evaluation.

[2] In some jurisdictions, a more conservative 5 REM limit is used.

isotope inventory in excess of Threshold II quantities given in Table 2[5,7,8] where the unmitigated consequence are associated with:

- not in Design Categories 4S or 4
- unmitigated2 off-site dose is in excess of 5 REM
- unmitigated2 on-site dose in excess of mandated limits for co-located workers
- the facility stores or processes radiological chemical or biological material or waste whose unmitigated release could have significant health or environmental effects at the site boundary.

3.4. Design Category 2

A SSC shall be placed in Design Category 2 if it is necessary to prevent or mitigate the consequences of design basis hazards to a facility which has a radiological isotope inventory in excess of Threshold I quantities given in Table 2[5,7,8], not in Categories 4S, 4 and 3 or can be categorized as Category III or IV from Table I-I from ASCE Std. 7-95.[6] attached hereto as Table III.

3.5. Design Category 1

A SSC shall be placed in Design Category 1 if categorized as Category II in Table I-I of ASCE Std. 7-95[6] attached hereto as Table III.

3.6. Design Category 0

An SSC shall be placed in Design Category 0 if categorized as Category I in Table I-I of ASCE Std. 7-95[6] attached hereto as Table III.

4. SUGGESTED DESIGN CATEGORIZATION OF NON-REACTOR NUCLEAR SAFETY RELATED AT NUCLEAR POWER PLANTS

The following is a suggested design categorization of nuclear safety related systems at NPP:

- reactor coolant and reactor safety systems — Category 4S
- new spent fuel and associated safety systems (less than 5 years old) — Category 4
- old spent fuel (older than 5 years) and associated safety systems — Category 3
- secondary confinement3 or containment4 of all nuclear waste or material in quantities in excess of Threshold II limits — Category 3
- primary confinement[4] or containment[4] of gaseous or liquid nuclear waste or material in quantities in excess of Threshold II limits — Category 3
- primary confinement[4] or containment[4] of solid nuclear waste or material in quantities in excess of Threshold II limits — Category 2
- primary and secondary confinement[4] or containment[4] of all nuclear waste or material in quantities less than Threshold II limits — Category 2
- primary and secondary confinement[4] or containment[4] of all nuclear waste or material in quantities less than Threshold I limits — Category 1

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[3] Containment is distinguished from confinement when the design basis for the component includes a design pressure in excess of 50kpa.

TABLE I. SUMMARY OF EXTERNAL NATURAL HAZARD DESIGN BASIS EVENT PROBABILITIES

	Performance Category	0	1	2	3 Probabilities	4 Per Year	4S
EARTHQUAKE	Mean Hazard Annual Prob. of Exceedance-New Facility	NO REQUIREMENT	2x10^{-3}	1x10^{-3}	5x10^{-4}	1x10^{-4}	1x10^{-5}
EARTHQUAKE	Mean Hazard Annual Prob. of Exceedence-Existing Facil.	NO REQUIREMENT	4x10^{-3}	2x10^{-2}	1x10^{-3}	2x10^{-4}	2x10^{-5}
WIND	Hazard Annual Prob. of Exceedance	NO REQUIREMENT	2x10^{-2}	2x10^{-2}	1x10^{-3}	1x10^{-4}	2x10^{-5}
WIND	Importance Factor	NO REQUIREMENT	1.0	1.07	1.0	1.0	1.0
WIND	Missile Criteria	NO REQUIREMENT	NA	NA	3 in. dia. std. steel pipe, 78 lb @ 50 mph (horiz.); max. height 75 ft, 35 mph (vert.)	3 in. dia. std. steel pipe, 78 lb @ 75 mph (horiz.); max. height 75 ft, 50 mph (vert.)	Tornado Missiles Govern
(1) TORNADO	Hazard Annual Prob. of Exceedance	NO REQUIREMENT	NA	NA	2x10^{-5}	2x10^{-6}	300 MPH E of 105° / 200 MPH W of 105°
(1) TORNADO	Importance Factor	NO REQUIREMENT	NA	NA	1 = 1.0	1 = 1.0	1 = 1.0
(1) TORNADO	Associated Pressure Drop	NO REQUIREMENT	NA	NA	40 psf @ 20 psf/sec	125 psf @ 50 psf/sec	Defined as a function of Wind Velocity
(1) TORNADO	Missile Criteria		NA	NA	3 in. dia. std. steel pipe, 78 lb @ 50 mph (horiz.); max. height 75 ft, 35 mph (vert.)	3 in. dia. std. steel pipe, 78 lb @ 75 mph (horiz.); max. height 75 ft, 50 mph (vert.) Automobile frontal area 20ft^2, weight 4000 lbs @ 35 mph horizontal	See Missile List Contained in Table 4 taken from USNRC SRP-3.5

TABLE I. (cont.)

	Performance Category	0	1	2	3 Probabilities	4 Per Year	4S
FLOOD	Mean Hazard Annual Prob. of Exceedence New Facility	NO REQUIREMENT	$2x10^{-3}$	$5x10^{-4}$	$1x10^{-4}$	$1x10^{-5}$	$1x10^{-5}$ See ANS Std. 2.8
PRECIPITATION	Rain and Snow		$2x10^{-3}$	$5x10^{-4}$	1×10^{-4}	1×10^{-5}	1×10^{-5}
	Surface Run-off						
	Roof Loads		$2x10^{-3}$	$5x10^{-4}$	1×10^{-4}	1×10^{-5}	1×10^{-5}
	Site Drainage		$4x10^{-3}$	$4x10^{-3}$	4×10^{-3}	4×10^{-3}	4×10^{-3}

(1) Tornado design requirements are limited to relatively few geographical areas in the world.

TABLE II. BASIC RADIONUCLIDE VALUES FOR UNKNOWN RADIONUCLIDES OR MIXTURES

Radioactive Contents	Threshold II[2]		Threshold I[2]	
	A_1[1]	A_2	A_1[1]	A_2
	TBq	TBq	TBq	TBq
Only beta or gamma emitting nuclides are known to be present	900	180	0.3	0.06
Only alpha emitting nuclides are known to be present	1800	270	0.6	0.09
No relevant data are available	1100	45	0.37	0.015

Notes and Definitions:

Bq = Becquerel
1.0Bq = $2.7x10^{-11}$ C_i
T = tera = 10^{12}
1.0TBq = $27C_i$
A_1 = Radionuclide quantities in special form[1]
A_2 = Radionuclide quantities not in special form
C_i = Curie

[1]Special form radioactive material is material in an indispersible solid form or in a robustly constructed sealed capsule.

[2]The quantities for Threshold II are related to Threshold I quantities by the multiplication factor 3000.[5,8]

TABLE III. LIST OF NRC TORNADO DESIGN MISSILES

		Fraction of Maximum Total Tornado Velocity
A.	Wood plank, 4 in. x 12 in. x 12 ft., weight 200 lb.	0.8
B.	Steel pipe, 3 in. diameter, schedule 40, 10 ft. long, weight 78 lb.	0.4
C.	Steel rod, 1 in. diameter x 3 ft. long, weight 8 lb.	0.6
D.	Steel pipe, 6 in. diameter, schedule 40, 15 ft. long, weight 285 lb.	0.4
E.	Steel pipe, 12 in. diameter, schedule 40, 15 ft. long, weight 743 lb.	0.4
F.	Utility pole, 13-1/2 in. diameter, 35 ft. long, weight 1,490 lb.	0.4
G.	Automobile, frontal area 20 ft^2, weight 4,000 lb.	0.2

REFERENCES

[1] 10 CFR 50 Appendix S, "Earthquake Engineering Criteria for Nuclear Power Plants," Federal Register (1997).

[2] NUREG-0800 Section 3.7, "Seismic Design," Rev. 2, Standard Review Plan, US Nuclear Regulatory Commission (1989).

[3] DOE O 420.1, "Facility Safety," US Department of Energy (1995).

[4] DOE Std. 1020-94, Ch. Notice 1 (1996), "Natural Phenomena Hazards Design and Evaluation Criteria for Department of Energy Facilities," US Department of Energy (1996).

[5] IAEA Safety Standards Series, "Regulation for the Safe Transport of Radioactive Material," No. St-1 International Atomic Energy Agency (1996).

[6] ASCE Std. 7-95, "Minimum Design Loads for Buildings and Other Structures," American Society of Civil Engineers (1995).

[7] DOE Std. 1027 Attachment 1, "Hazard and Accident Analysis Techniques for Compliance with DOE Order 5480.23, Nuclear Safety Analysis Reports," US Department of Energy (1992).

[8] Regulatory Guide 7.11, "Fracture Toughness Criteria of Base Material for Verities Steel Shipping Cask Containment Vessels with Maximum Wall Thickness (0.1m)," US Nuclear Regulatory Commission (1991).

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DRAFT REGULATORY GUIDE

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DRAFT REGULATORY GUIDE DG-1100

(Proposed Revision 2 to Regulatory Guide 1.143)

DESIGN GUIDANCE FOR RADIOACTIVE WASTE MANAGEMENT SYSTEMS, STRUCTURES, AND COMPONENTS INSTALLED IN LIGHT-WATER-COOLED NUCLEAR POWER PLANTS

A. INTRODUCTION

This regulatory guide is being revised to provide guidance to licensees and applicants on methods acceptable to the staff for complying with the NRC's regulations in the design, construction, installation, and testing of radioactive waste management facilities, and the structures, systems, and components in light-water-reactor nuclear power plants.

In 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," § 50.34, "Contents of Applications; Technical Information," requires that each application for a construction permit include a preliminary safety analysis report. Part of the information required is related to quality assurance and the preliminary design of the facility, including, among other things, the principal design criteria for the facility. Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50 establishes overall quality assurance requirements for structures, systems, and components important to safety. Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50 establishes minimum requirements for the principal design criteria for light-water-cooled nuclear power plants.

Criterion 1, "Quality Standards and Records," of Appendix A requires that structures, systems, and components important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance to safety of the safety function to be performed and that a quality assurance program be established and implemented in order to provide adequate assurance that these structures, systems, and components will satisfactorily perform their safety function.

Criterion 2, "Design Bases for Protection Against Natural Phenomena," of Appendix A requires, among other things, that structures, systems, and components important to safety be designed to

This regulatory guide is being issued in draft form to involve the public in the early stages of the development of a regulatory position in this area. It has not received complete staff review and does not represent an official NRC staff position.

Public comments are being solicited on the draft guide (including any implementation schedule) and its associated regulatory analysis or value/impact statement. Comments should be accompanied by appropriate supporting data. Written comments may be submitted to the Rules and Directives Branch, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001. Comments may be submitted electronically or downloaded through the NRC's interactive web site at <WWW.NRC.GOV> through Rulemaking. Copies of comments received may be examined at the NRC Public Document Room, 2120 L Street NW., Washington, DC. Comments will be most helpful if received by November 20, 2000.

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withstand the effects of natural phenomena such as earthquakes, extreme winds, tornados, or flooding without loss of capability to perform their safety functions. The design bases for these structures, systems, and components are to reflect the importance of the safety functions to be performed. Appendix S,"Earthquake Engineering Criteria for Nuclear Power Plants," of 10 CFR Part 50 states general design requirements for the implementation of General Design Criterion 2. Criterion 60, "Control of Releases of Radioactive Materials to the Environment," of Appendix A requires that the nuclear power unit design include suitable means to control the release of radioactive materials in gaseous and liquid effluents and to handle radioactive solid waste produced during normal reactor operation, including anticipated operational and external man-induced and design basis accident occurrences.

This regulatory guide is being revised to provide design guidance acceptable to the NRC staff on natural phenomena hazards, internal and external man-induced hazards, and quality group classification and quality assurance provisions for radioactive waste management systems, structures, and components.[1] Further, it describes provisions for mitigating design basis accidents, controlling releases of liquids containing radioactive materials, e.g., spills or tank overflows, from all plant systems outside reactor containment.

Licensees and applicants may propose means other than those specified by the provisions of the Regulatory Position of this for meeting applicable regulations. No new requirements are being imposed by this draft regulatory guide. Implementation of this guidance by licensees will be on a strictly voluntary basis.

Regulatory guides are issued to describe to the public methods acceptable to the NRC staff for implementing specific parts of the NRC's regulations, to explain techniques used by the staff in evaluating specific problems or postulated accidents, and to provide guidance to applicants. Regulatory guides are not substitutes for regulations, and compliance with regulatory guides is not required. Regulatory guides are issued in draft form for public comment to involve the public in developing Regulatory Positions. Draft regulatory guides have not received complete staff review; they therefore do not represent official NRC staff positions.

The information collections contained in this draft regulatory guide are covered by the requirements in 10 CFR Part 50, which were approved by the Office Management and Budget, approval number 3150-0011. If a means used to impose an information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.

B. DISCUSSION

One aspect of nuclear power plant operation is the control and management of liquid, gaseous, and solid radioactive waste[2] (radwaste) generated as a byproduct of nuclear power.

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[1] Adams et al, "Re-evaluation of Regulatory Guidance Provided in Regulatory Guides 1.142 and 1.142," NUREG/CR-5733, August 1999. Copies are available at current rates from the U.S. Government Printing Office, P.O. Box 37082, Washington, DC 20402-9328 (telephone (202)512-1800); or from the National Technical Information Service by writing NTIS at 5285 Port Royal Road, Springfield, VA 22161; (telephone (703)487-4650; <http://www.ntis.gov/ordernow>. Copies are available for inspection or copying for a fee from the NRC Public Document Room at 2120 L Street NW., Washington, DC; the PDR's mailing address is Mail Stop LL-6, Washington, DC 20555; telephone (202)634-3273 or (800)397-4209; fax (202)634-3343; email is PDR@NRC.GOV.T.M.

[9] Radioactive waste, as used in this guide, means those liquids, gases or solids containing radioactive materials that by design or operating practice will be processed prior to final disposition.

The purpose of this guide is to provide information and criteria that will provide reasonable assurance that components and structures used in the radioactive waste management and steam generator blowdown systems are designed, constructed, installed, and tested on a level commensurate with the need to protect the health and safety of the public and plant operating personnel. It sets forth minimum staff recommendations and is not intended to prohibit the implementation of more rigorous design considerations, codes, standards, or quality assurance measures.

ANSI/ANS Standards 55.1-1992, “Solid Radioactive Waste Processing System for Light Water Cooled Reactor Plants,”[3] 55.4-1993, “Gaseous Radioactive Waste Processing Systems for Light Water Plants,”[3] and 55.6-1993, “Liquid Radioactive Waste Processing Systems for Light Water Reactor Plants,”[3] have been reviewed for applicability to this guide. These ANSI/ANS Standards provide a wider range of guidance than that provided in Section 11.0 of NUREG-0800, “USNRC Standard Review Plan.”[4] As appropriate, guidance from the ANSI/ANS Standards has been incorporated by reference.

For the purposes of this guide, the radwaste systems are considered to begin at the interface valves in each line from other systems provided for collecting wastes that may contain radioactive materials and to include related instrumentation and control systems. The radwaste system terminates at the point of controlled discharged to the environment, at the point of recycle to the primary or secondary water system storage tanks, or at the point of storage of packaged solid wastes prior to shipment offsite to a licensed burial ground.

The steam generator blowdown system begins at, but does not include, the outermost containment isolation valve on the blowdown line. It terminates at the point of controlled discharge to the environment, at that point of interface with other liquid systems, or at the point of recycle back to the secondary system. For design purposes, portions of radwaste systems that interface with other systems are considered to be in the system with more rigorous requirements.

Except as noted, this guide does not apply to the reactor water cleanup system, the condensate cleanup system, the chemical and volume control system, the reactor coolant and auxilliary building equipment drain tanks, the sumps and floor drains provided for collecting liquid wastes, the boron recovery system, equipment used to prepare solid waste solidification agents, the building ventilation systems (heating, ventilating, and air conditioning), instrumentation and sampling systems beyond the first root valve, or the chemical fume hood exhaust systems. In addition, this guide does not apply to the main condenser circulating or component cooling water systems, or the spent fuel handing and storage systems, or the fuel pool water cleanup system.

The design and construction of radioactive waste management and steam generator blowdown systems should provide assurance that radiation exposures to operating personnel and to the general public are as low as is reasonably achievable. One aspect of this consideration is ensuring that these systems are designed to quality standards that enhance system reliability, operability, and availability. In developing this design guidance, the NRC staff has considered designs and concepts submitted in license applications and resulting

[9] Copies may be obtained from the American Nuclear Society, 555 North Kensington Avenue, La Grange Park, Illinois 60525.

[9] Copies of sections of NUREG 0800 are available by email to DISTRIBUTION@NRC.GOV or by fax to (301)415-2289.

operating system histories. It has also been guided by industry practices and the cost of design features, taking into account the potential impact on the health and safety of operating personnel and the general public.

C. REGULATORY POSITION

1. SYSTEMS HANDLING RADIOACTIVE MATERIALS IN LIQUIDS

1.1 Liquid Radwaste Treatment System

The liquid radwaste treatment system, including the steam generator blowdown system, downstream of the outer most containment isolation valve should meet the following criteria:

1.1.1 The structures, systems, and components (SSCs) of the liquid radwaste treatment system should be designed and tested to requirements set forth in the codes and standards listed in Table 1, supplemented by Regulatory Positions 1.1.2 and 1.1.3 of this guide.

1.1.2 Materials for pressure-retaining components, excluding HVAC duct and fire protection piping, should conform to the requirements of the specifications for materials listed in Section II of the ASME Boiler and Pressure Vessel Code,[5] except that malleable, wrought, or cast iron materials and plastic pipe should not be used. Materials should be compatible with the chemical, physical, and radioactive environment of specific applications during normal conditions and anticipated operational occurrences. Manufacturers' material certificates of compliance with material specifications such as those contained in the codes referenced in Table 1 may be provided in lieu of certified material test reports.

1.1.3 Foundations and walls of structures that house the liquid radwaste system should be designed to the natural phenomena and internal and external man-induced hazards criteria described in Regulatory Position 6 of this guide to a height sufficient to contain the maximum liquid inventory expected to be in the building.

1.2 SSCs Outside Containment that Contain Radioactive Liquids

All SSCs located outside of the reactor containment and containing radioactive materials in liquid form should be classified as described in Regulatory Position 5 and designed to the criteria put forth in Regulatory Position 6. In addition, any such component should be designed to prevent uncontrolled releases of radioactive materials caused by spillage in buildings or from outdoor components. The following design features should be included for such components and should meet the criteria contained in Sections 5.2, 5.3, and 5.4 of ANSI/ANS 55.1-1191.

1.2.1 All tanks inside and outside the plant, including the condensate storage tanks, should have provisions to monitor liquid levels. Designated high-liquid-level conditions should actuate alarms both locally and in the control room.

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[9] Copies may be obtained from the American Society of Mechanical Engineers, United Engineering Center, 345 East 47th Street, New York, NY 10017.

1.2.2 All radwaste tanks, overflows, drains, and sample lines should be routed to the liquid radwaste treatment system.[6]

1.2.3 Indoor radwaste tanks should have curbs or elevated thresholds with floor drains routed to the liquid radwaste treatment system.[6]

1.2.4 The design should include provisions to prevent leakage from entering unmonitored and nonradioactive systems and ductwork in the area.

1.2.5 Outdoor tanks should have a dike or retention pond capable of preventing runoff in the event of a tank overflow and should have provisions for sampling collected liquids and routing them to the liquid radwaste treatment system.

2. GASEOUS RADWASTE SYSTEMS

The gaseous radwaste treatment system[7] should meet the following criteria:

2.1 The SSCs of the gaseous radwaste treatment system should be designed and tested to requirements set forth in the codes and standards listed in Table 1 supplemented by Regulatory Positions 2.2 and 2.3 of this guide.

2.2 Materials for pressure-retaining components, excluding HVAC duct and fire protection piping, should conform to the requirements of the specifications for materials listed in Section II of the ASME Boiler and Pressure Vessel Code, except that malleable, wrought, or cast iron materials and plastic pipe should not be used. Materials should be compatible with the chemical, physical, and radioactive environment of specific applications during normal conditions and anticipated operational occurrences. If the potential for an explosive mixture of hydrogen and oxygen exists, adequate provisions should be made to preclude buildup of explosive mixtures, or the system should be designed to withstand the effects of an explosion. Manufacturers' material certificates of compliance with material specifications such as those contained in the codes referenced in Table 1 may be provided in lieu of certified materials test reports.

2.3 The portions of the gaseous radwaste treatment system that are intended to store or delay the release of gaseous radioactive waste, including portions of structures housing these systems, should be classified as described in Regulatory Position 5 and designed to the criteria of Regulatory Position 6.

3. SOLID RADWASTE SYSTEM

The solid radwaste system consists of slurry waste collection and settling tanks, spent resin storage tanks, phase separators, and components and subsystems used to solidify radwastes prior to offsite shipment. The solid radwaste handling and treatment system should meet the following criteria.

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[9]Retention by an intermediate sump or drain tank that is designed for handling radioactive materials and that has provisions for routing to the liquid radwaste system is acceptable.

[9] For a BWR this includes the system provided for treatment of normal offgas releases from the main condenser vacuum system beginning at the point of discharge from the condenser air removal equipment; for a PWR this includes the sytem provided for the treatment of gases stripped from the primary coolant.

3.1 The SSCs of the solid radwaste treatment system should be designed and tested to the requirements set forth in the codes and standards listed in Table 1 supplemented by Regulatory Positions 3.2 and 3.3 of this guide.

3.2 Materials for pressure-retaining components, excluding HVAC duct and fire protection piping, should conform to the requirements of the specifications for materials listed in Section II of the ASME Boiler and Pressure Vessel Code, except that malleable, wrought, or cast iron materials and plastic pipe should not be used. Materials should be compatible with the chemical, physical, and radioactive environment of specific applications during normal conditions and anticipated operational occurrences. Manufacturers' material certificates of compliance with material specifications such as those contained in the codes referenced in Table 1 may be provided in lieu of certified materials test reports.

3.3 Foundations and adjacent walls of structures that house the solid radwaste system should be designed to the natural phenomena and internal and external man-induced hazards criteria given in Regulatory Position 6 of this guide to a height sufficient to contain the maximum liquid inventory expected to be in the building.

3.4 Equipment and components used to collect, process, or store solid radwastes need not be designed to the seismic criteria in Regulatory Position 6 of this guide.

4. ADDITIONAL DESIGN, CONSTRUCTION, AND TESTING CRITERIA

In addition to the requirements inherent in the codes and standards listed in Table 1, the following criteria, as a minimum, should be applicable to SSCs listed in Regulatory Position 6 of this guideline.

4.1 Radioactive waste management SSCs should be designed to control leakage and facilitate access, operation, inspection, testing, and maintenance in order to maintain radiation exposures to operating and maintenance personnel as low as is reasonably achievable. Regulatory Guide 8.8, "Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will Be As Low As Is Reasonably Achievable," provides guidelines that are acceptable to the NRC staff on this subject.

4.2 The quality assurance provisions described in Regulatory Position 7 of this guide should be applied.

4.3 Pressure-retaining components of process systems should use welded construction to the maximum practicable extent. Process systems include the first root valve on sample and instrument lines. Flanged joints or suitable rapid-disconnect fittings should be used only where maintenance or operational requirements clearly indicate such construction is preferable. Screwed connections in which threads provide the only seal should not be used except for instrumentation and cast pump body drain and vent connections where welded connections are not suitable. Process lines should not be less than 3/4 inch (nominal). Screwed connections backed up by seal welding, mechanical joints, or socket welding may be used on lines 3/4 inches or larger but less than 2-1/2 inches. For lines 2–1/2 inches and above, pipe welds should be of the butt-joint type. Nonconsumable backing rings should not be used in lines carrying resins or other particulate material. All welding constituting the pressure boundary of pressure-retaining components should be performed in accordance with ASME Boiler and Pressure Vessel Code Section IX.

4.4 Piping systems should be hydrostatically tested in their entirety except (1) at atmospheric tanks where no isolation valves exist, (2) where such testing would damage equipment, and (3) where such testing could seriously interfere with other system or component testing. In the case of (2) and (3), pneumatic testing should be performed. Pressure testing should be performed on as large a portion of the in-place systems as practicable. Testing of piping systems should be performed in accordance with applicable ASME or ANSI codes listed in Table 1.

4.5 In-service inspection and testing provisions should be incorporated to enable periodic evaluation of the operability and required functional performance of active components of the system.

5. CLASSIFICATION OF RADWASTE SYSTEMS FOR DESIGN PURPOSE

Three safety classes, or classifications, are established for radwaste management facilities: RW-IIa (High Hazard), RW-IIb (Hazardous), and RW-IIc (Non-Safety). RW-IIa is the most stringent class and RW-IIc is the least stringent. These classifications were developed primarily for natural phenomena and man-induced hazard design. This safety classification is applied to the SSC as follows.

5.1 For a given structure housing radwaste processing systems or components, if the total design basis unmitigated release (considering the maximum inventory) at the boundary of the unprotected area is greater than 500 millirem per year or the maximum unmitigated exposure to site personnel within the protected area is greater than 5.0 rem per year, the external structures are classified as RW-IIa.

5.2 For a given structure housing radwaste processing systems or components, if the total design basis unmitigated radiological release (considering the maximum inventory) at the boundary of the unprotected area is less than 500 millirem per year and the maximum unmitigated exposure to site personnel within the protected area is less than 5 rem per year, the external structure is classified as RE-IIb.

5.3 Any systems or components in a RW-IIa facility (see Regulatory Position 5.1) that store, process, or handle radioactive waste in excess of the A_1 quantities given in Appendix A, "Determination of A_1 and A_2," to 10 CFR Part 71, "Packaging and Transportation of Radioactive Material," are classified as RW-IIa. These systems or components that process radioactive waste in excess of the A_2 quantities but less than the A_1 quantities given in Appendix A to 10 CFR Part 71 are classified as RW-IIb. All other components are classified as RW-IIc. This classification may be modified for specific radwaste components as discussed in the subsections that follow.

5.4 Any systems or components in a RW-IIb structure (see Regulatory Position 5.2) that are used to store or process specified radioactive waste in excess of the A_1 quantities given in Appendix A to 10 CFR Part 71 are classified as RW-IIb. All other systems or components are classified as RW-IIc.

The unprotected area boundary mentioned in Regulatory Position 5.1 is shown in Figure 1. A flowchart of the Safety Classification Process is shown in Figure 2.

6. NATURAL PHENOMENA AND MAN-INDUCED HAZARDS DESIGN FOR RADWASTE MANAGEMENT SYSTEMS AND STRUCTURES

6.1 General Design Criteria

Solid, liquid, and gaseous radwaste SSCs described in Regulatory Positions 1, 2, and 3 for natural phenomena and internal and external man-induced hazards should be evaluated as put forth in this position.

6.1.1 The natural phenomena and internal and external man-induced hazards demand definition is as given in Table 2.

6.1.2 The natural phenomena and internal and external man-induced hazards design load combinations are as given in Table 3.

6.1.3 The natural phenomena and internal and external man-induced hazards should meet capacity criteria as defined in Table 4.

The acceptability evaluation should be based on the requirements of the codes and standards given in Table 1, using the capacity criteria in Table 4.

6.2 Buildings Housing Radwaste Systems

6.2.1 Regardless of its safety classification, the foundation and walls up to the spill height of the building housing the radwaste systems should be classified RW-IIa and they are to be designed to the criteria of Tables 1, 2, 3, and 4.

For classifications RW-IIb and RW-IIc, the staff recommends that all SSCs be designed at least for local building code seismic base shear requirements. In the absence of such local building criteria, it is recommended that the guidance of Volume 2 of the Standard Uniform Building Code 1997,[8] and American Society of Civil Engineers ASCE 7-957, "Minimum Design Loads for Buildings and Other Structures,"[9] be used as noted in Table 2 of this regulatory guide.

6.2.2 In lieu of the criteria and procedures referenced in Regulatory Position 6, optional shield structures constructed around and supporting the radwaste systems may be erected to protect the radwaste systems from effects of failure of the housing structure. If this option is adopted, the procedure described in Regulatory Position 6.2.1 need only be applied to the shield structures.

7. QUALITY ASSURANCE FOR RADWASTE MANAGEMENT SYSTEMS

Since the impact of these systems on safety is limited, the extent of control required by Appendix B to 10 CFR Part 50 is similarly limited. To ensure that systems will perform their intended functions, a quality assurance program sufficient to ensure that all design,

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[9] Copies may be obtained from the International Conference of Building Officials, 5360 Workman Mill Road, Whittier, CA 90601-2798.

[9] Copies may be obtained from the American Society of Civil Engineers, 345 East 47th Street, New York, NY 10017-2398.

construction, and testing provisions are met should be established and documented. A quality assurance program acceptable to the NRC staff is presented in ANSI N199-1993/ANS-55.2, "Liquid Radioactive Waste Processing System for Pressurized Water Reactor Plants."

Section 4.3, "Quality Assurance," of ANSI N199-1993/ANS 55.2 provides quality assurance guidance that is acceptable to the NRC staff for the system designer and procurer and for the system constructor. The design, procurement, fabrication, and construction activities should conform to the quality control provisions of the codes and standards specified in Table 1 of this draft guide. In addition, or where not covered by the referenced codes and standards, sufficient records should be maintained to furnish evidence that quality assurance measures are being implemented. The records should include results of reviews and inspections and should be identifiable and retrievable.

D. IMPLEMENTATION

The purpose of this section is to provide information to licensees and applicants regarding the NRC staff's plans for using this regulatory guide.

This proposed revision has been released to encourage public participation in its development. Except in those cases in which the applicant or licensee proposes an acceptable alternative method for complying with specified portions of the NRC's regulations, the method described in the active guide reflecting public comments will be used in the evaluation of a licensee's or applicant's design, construction, installation, and testing of radioactive waste management facilities, and in the evaluation of structures, systems, and components in light-water-cooled nuclear power plants.

Table 1 - Codes and Standards for the Design of SSC in Radwaste Facilities[1]

Component	Design and Construction	Materials	Welding	Inspection and Testing
Structures - Concrete	ACI-318 or ACI 349[(2)(3)]	ACI-318 or ACI 349	ACI-318 or ACI 349	ACI-318 or ACI 349
Structures-Steel (Hot Rolled)	AISC-ASD or AISC LFRD or AISC N-690(S327) [(2)(4)]	ASTM-A36	AWS-D1.1	AISC Standards and AWS Standards
Structures-Steel (Cold Formed)	AISI-CFSDM	ASTM-A500	AWS-D1.3, D9.1	AISC Standards and AWS Standards
Piping & Valves	ANSI/ASME B31.3 [(5)(6)]	ASME-Sec. II[(7)]	ASME, Sec. IX	ANSI/ASME B31.3
Atmospheric Tanks	API-650	ASME Sec. II	ASME, Sec. IX	API-620
Tanks (0-15 psig)	API-620	ASME Sec. II	ASME, Sec. IX	API-650
Pressure Vessels and Tanks (>15 psig)	ASME BPVC Div. 1 or Div. 2	ASME Sec. II	ASME, Sec. IX	ASME Section VIII, Div. 1 or 2
Pumps	API-610; API-674; API-675; ASME BPVC Section VIII, Div. 1 or Div. 2	ASTM or ASME Sec. II	ASME, Sec. IX	ASME BPVC Code Section III, Class 3[(8)]
Heat Exchangers	TEMA STD, 7th Edition; ASME BPVC Section VIII Div. 1 or Div. 2	ASTM-B359 or ASME Sec. II	ASME, Sec. IX	ASME Section VIII, Div. 1 or 2
HVAC Systems	SMACNA Stds. [(9)(6)]	ASTM	AWS-D1.1, D1.3, D9.1	SMACNA Stds
Conduit and Cable Trays	ANSI/NEMA Standards	ASTM/NEMA	AWS-D1.1, D1.3, D9.1,	ANSI/NEMA Stds.
Fire Protection Systems	NFPA-13[(10)(6)]; NFPA-14	ASTM-A795	AWS-D1.1, D1.3, D9.1, D10.9	NFPA-13

Footnotes for Table 1:

(1) For a comprehensive lists of codes and standards referenced in Tables 1 4 see Appendix A to this regulatory guide.
(2) Applicable to structure enclosing or supporting pressurized gaseous waste or liquid waste systems up to spill height. Also applicable to solid waste facility foundations slab and connected wall or column sections up to a height of 10.0 feet.
(3) Appropriate load combinations and capacity criteria for component designs are specified in Table 3 of this regulatory guide.
(4) Class RW-IIa Structures are to use ACI-349 and/or AISC N-690(S327) as applicable.

(5) Class RW-IIa and RW-IIb Piping Systems are to be designed as category "M" systems.

(6) Classes RW-IIa, RW-IIb, and RW-IIc are discussed in Regulatory Position 5 of this regulatory guide.

(7) ASME BPVC Section II required for Pressure Retaining Components.

(8) ASME Code Stamp, material traceability, and the quality assurance criteria of ASME BPVC, Section III, Div. 1, Article NCA are not required. Therefore, these components are not classified as ASME Code Class 3.

(9) Class RW-IIa and RW-IIb HVAC systems are to use SMACNA "Seismic Restraint Manual Guides for Mechanical Systems."

(10) Class RW-IIa and RW-IIb Fire Protection Systems are to be designed to NFPA-13, Section 4-14.4.3.

Table 2 - Natural Phenomena and External Man-Induced Hazard Design Criteria for Safety Classification			
Loading	**Classification**		
	RW-IIa (High Hazard)	**RW-IIb (Hazardous)**	**RW-IIc (Non-Safety)**
Earthquake	OBE or 1/2 SSE	ASCE 7-95, Category III[1] or UBC 97, Category 2[2]	ASCE 7-95, Category II[1] UBC-97, Category 4[2]
Wind	ASCE 7-95, Category III[1]	ASCE 7-95, Category III[1]	ASCE 7-95, Category II[1]
Tornado	ANS 2.3 at a Probability of 1 x 10^{-5} /yr or three-fifths of Criteria in Regulatory Guide 1.76, Table 1.	ASCE 7-95, Category III[1]	ASCE 7-95, Category II[1]
Tornado Missile **from SRP Section 3.5**	A. 75 lbs, 3 in. nominal diameter sch. 40 pipe. Maximum velocity 0.4 x max. wind speed horizontal and 0.28 times max. wind speed vertical direction.[3] B. Automobile wt. 4000 lbs with frontal area of 20.0 sq. ft. traveling horizontally at 0.2 times maximum wind speed horizontally and 0.14 times maximum wind speed up to a height of 35 ft above grade[4].	Not Required	Not Required
Flood	Regulatory Guide 1.59, one-half of the PMF.(5).	ASCE 7-95, Category III[1]	ASCE 7-95, Category II[1]
Precipitation [6] (Rain, Snow)	ANS 2.6 at Probability of 1 x 10^{-3}/yr or Regulatory Guide 1.59, one-half precipitation specific for the PMF[5]	ASCE 7-95, Category III[1]	ASCE 7-95, Category II[1]
Accidental Explosion Fixed Facility	To Be Evaluated on a Case-by-Case Basis, Plant-Specific Definition	Not Required	Not Required
Accidental Explosion Transportation Vehicle	See Regulatory Guide 1.91	Not Required	Not Required
Malevolent Vehicle Assault	Regulatory Guide 5.68 or Plant Specific Definition	Not Required	Not Required
Small Aircraft Crash	Plant-Specific Definition	Not Required	Not Required

Footnotes for Table 2:

(1) ASCE 7-95, Table 1-1
(2) UBC-97, Table 16-k
(3) Penetrating-type missile.
(4) Impact-type missile.
(5) PMF = Probable Maximum Flood.
(4) Resistance to lightening strike should also be included in the design.

Table 3 - Design Load Combinations				
System, Structure, Component (SSC)	**Service Levels**	**SSC Safety Class**		
		RW - IIa	**RW- IIb**	**RW- IIc**
External Structures (Concrete, Steel, Component Support Structures[1]) External Conduits and Cable Trays	A (Normal)	D + L + To	D + L + To	D + L + To
	B (Severe; Upset)	D + L + Tb D + L + To + Eo D + L + To + W + R D + L + To + F	D + L + Tb D + L + To + E’o D + L + To + W + R D + L + To + F	D + L + Tb D + L + To + E”o D + L + To + W + R D + L + To + F
	D (Abnormal Extreme; Faulted)	$D + L + To + W_t$ $D + L + To + V_m$ $D + L + To + A_c$ $D + L + T_a + A_D$ $D + L + T_a + A$	N/R	N/R
Internal Structures (Concrete, Steel Component Support Structures[1]) Internal Conduit and Cable Trays	A (Normal)	D + L + To	D + L + To	D + L + To
	B (Severe, Upset)	$D + L + T_b$ $D + L + T_o + E_o$ $D + L + T_o + F$	$D + L + T_b$ $D + L + T_o + E_o$ $D + L + T_o + F$	$D + L + T_b$ $D + L + T_o + E_o$ $D + L + T_o + F$
	D (Abnormal Extreme; Faulted)	$D + L + T_a + A_D$ $D + L + T_a + A$	N/R	N/R
Pressure Retaining Components[2] (Piping, Valves, Pressure Vessels, Atmosphere, Tanks, 0-15 psig Tanks, Pumps Heat Exchangers) HVAC Systems Fire Protection Systems	A (Normal)	$P_D + D + D_m$ T_o	$P_d + D + D_M$ T_o	$P_D + D + D_m$ T_o
	B (Severe, Upset)	$P_o + D + D_m + E_o$ $P_o + D + D_m + W + R$ $P + D + D_m + F$ T_b	$P_o + D + D_m + E_o$ $P_o + D + D_m + W + R$ $P + D + D_m + F$ T_b	$P_o + D + D_m + E_o$ $P_o + D + D_m + W + R$ $P + D + D_m + F$ T_b
	D (Abnormal, Extreme Faulted)	$P + D + D_m + W_t$ $P_o + D + D_m + Y_m$ $P_o + D + D_m + A_c$ $P_a + D + D_m + A_D$ $P_a + D + D_m + A$	N/R	N/R

Nomenclature:

D	=	Dead Loads
L	=	Live loads
T_o	=	Normal Operating Thermal Expansion Loads
T_b	=	Upset Thermal Expansion Loads
T_a	=	Accident Thermal Loads

E_o	=	OBE or ½ SSE Seismic Loads
E'_o	=	Seismic Loads per Table 2 For RW-IIb Components
E''_o	=	Seismic Loads per Table 2 For RW-IIc Components
W	=	Wind Load Including Missile Effects
R	=	Precipitation Loads (Rain, Snow)
F	=	Flood Loadings
W_t	=	Tornado Loads Including Missile Effects
V_m	=	Malevolent Vehicle Assault Loads
A_c	=	Aircraft Crash Loads
A_D	=	Design Basis Accident Loads
A	=	Other Accident Loads
P_D	=	Design Pressure
P_b	=	Maximum Upset Pressure
P_o	=	Normal Operating Pressure
P_a	=	Applicable Accident Pressure
D_M	=	Design Mechanical Loads

Footnotes:

(1) Components support structures include supporting elements for piping, tanks, vessels pumps, heat exchangers, conduits, cable trays, HVAC systems, fire protection systems, etc.

(2) For most pressure-retaining components, primary and secondary stresses are evaluated separately to separate criteria. The design code of record is the controlling document in the establishment of the primary and secondary stress combination and evaluation methods.

Table 4 - SSC Design Capacity Criteria				
Code or Standard	**Service Level**	**Capacity Criteria**		
		RW-IIa	**RW-IIb**	**RW-IIc**
ACI-349	A, B, D	Load Factors and Capacity Criteria per ACI-349 as modified by Regulatory Guide 1.143	N/A	N/A
ACI-318	A, B, D	Load Factors and capacity criteria per ACI-349 as modified by Regulatory Guide 1.142. All other design per ACI-318 criteria	Load Factors and capacity criteria per ACI-349 as modified by Regulatory Guide 1.142. All other design per ACI-318 criteria	Load Factors and capacity criteria per ACI-349 as modified by Guide 1.142. All other design per ACI-318 criteria
AISC-N690	A	Capacity criteria Table Q 1.5.7.1 for normal loads.	Capacity criteria Table Q 1.5.7.1 for normal loads.	Capacity criteria Table Q 1.5.7.1 for normal loads
	B	Capacity criteria 1.33 times that for Level A loads	Capacity criteria 1.33 times that for Level A loads	Capacity criteria 1.33 times that for Level A loads
	D	Capacity criteria per Table Q.1.5.7.1 for Abnormal Extreme Loads	N/R	N/R
AISC-ASD	A	Capacity Criteria per "Specification for Structural Steel Buildings Stress Design and Plastic Design, Part 1"	Capacity Criteria per "Specification for Structural Steel Buildings Stress Design and Plastic Design, Part 1"	Capacity Criteria per "Specification for Structural Steel Buildings Stress Design and Plastic Design, Part 1"
	B	Capacity Criteria 1.33 times that for Level A loads.	Capacity Criteria 1.33 times that for level A loads.	Capacity Criteria 1.33 times that for level A loads.
	D	Capacity Criteria per "Specification for Structural Steel Buildings Stress Design and Plastic Design, Part 2".	N/R	N/R

Table 4 - SSC Design Capacity Criteria (continued)				
Code or Standard	**Service Level**	**Capacity Criteria**		
		RW-IIa	**RW-IIb**	**RW-IIc**
AISI-CFSDM	A	Capacity criteria per “Specification for the Design of Cold Formed Steel Structural Members”	Capacity criteria per “Specification for the Design of Cold Formed Steel Structural Members”	Capacity criteria per “Specification for the Design of Cold Formed Steel Structural Members”
	B	Capacity criteria 1.33 times Level A	Capacity criteria 1.33 times Level A	Capacity criteria 1.33 times Level A
	D	Capacity criteria 1.6 times Level A	N/R	N/R
ANSI/ASME B31.3	A	B31.3 Design Load Capacities	B31.3 Design Load Capacities	B31.3 Design Load Capacities
	B	B31.3 Occasional Load Capacities	B31.3 Occasional Load Capacities	B31.3 Occasional Load Capacities
	D	1.8 Times B31.3 Occasional Load Capacities	N/R	N/R
ASME BPVC, Section VIII, Div. 1 or Div. 2	A	ASME BPVC, Section VIII, Div. 1 or Div. 2 Design Capacities	ASME BPVC, Section VIII, Div. 1 or Div. 2 Design Capacities	ASME BPVC, Section VIII, Div. 1 or Div. 2 Design Capacities
	B	Capacity criteria 1.2 Times Level A criteria	Capacity criteria 1.2 Times Level A criteria	Capacity criteria 1.2 Times Level A criteria
	D	Capacity criteria 1.8 times Level A criteria	N/R	N/R
SMACNA Stds.(1)	A	SMACNA Design Criteria	SMACNA Design Criteria	SMACNA Design Criteria
	B	SMACNA Design Criteria	SMACNA Design Criteria	SMACNA Design Criteria
	D	(1) Duct support members to meet capacity criteria for AISI-DFSDM or AISC-ASD for Level D Loads (2) Ducting stresses to be less than the material yield stress and shall be limited to 2/3 critical buckling	N/R	N/R
NFPA-13(1)	A	NFPA Design Criteria	NFPA Design Criteria	NFPA Design Criteria
	B	NFPA Design Criteria for Earthquake and Wind Loads	NFPA Design Criteria for Earthquake and Wind Loads	NFPA Design Criteria for Earthquake and Wind Loads

Table 4 - SSC Design Capacity Criteria (continued)				
Code or Standard	**Service Level**	**Capacity Criteria**		
		RW-IIa	**RW-IIb**	**RW-IIc**
	D	(3) Support members to meet capacity criteria for AISI-CFSDM or AISC-ASD for Level D Loads (4) Piping Stresses to meet the B31.3 Level D Capacity Criteria	N/R	N/R
ANSI/NEMA STDS (Cable Trays/Conduit)	A	ANSI/NEMA Design Criteria for Normal Loads	ANSI/NEMA Design Criteria for Normal Loads	ANSI/NEMA Design Criteria for Normal Loads
	B	ANSI/NEMA Design Criteria for Wind and Seismic Loads	ANSI/NEMA Design Criteria for Wind and Seismic Loads	ANSI/NEMA Design Criteria for Wind and Seismic Loads
	D	(5) Support members to meet capacity criteria for AISI-CFSDM or AISC-ASD for Level D Load (6) Trays and members to meet the capacity criteria for AISI-CFSDM for Level D Loads	N/R	N/R
Pumps (API Series STDS)	A	Applicable API Standards for Design Criteria	Applicable API Standards for Design Criteria	Applicable API Standards for Design Criteria
	B	ASME QME-1 1997	ASME QME-1 1997	ASME QME-1 1997
	D	ASME QME-1 1997	N/R	N/R
API-620/650 (Tanks)	A	API Design Capacity Criteria	API Design Capacity Criteria	API Design Capacity Criteria
	B	Capacity Criteria per ASME-BPVC - Section III, NC-3800, NC-3900 for Level B loads. All other Design per API Criteria.	Capacity Criteria per ASME-BPVC - Section III, NC-3800, NC-3900 for Level B loads. All other Design per API Criteria.	Capacity Criteria per ASME-BPVC - Section III, NC-3800, NC-3900 for Level B loads. All other Design per API Criteria.
	D	Capacity Criteria per ASME-BPVC - Section III, NC-3800, NC-3900 Level D Loads. All other Design per API Criteria.	N/R	N/R

Footnotes for Table 4:

(1) For Level A and B Loads, the Design Criteria is primarily a “design by rule” approach versus a specific analysis criteria.
N/A= Not Applicable
N/R = Not Required

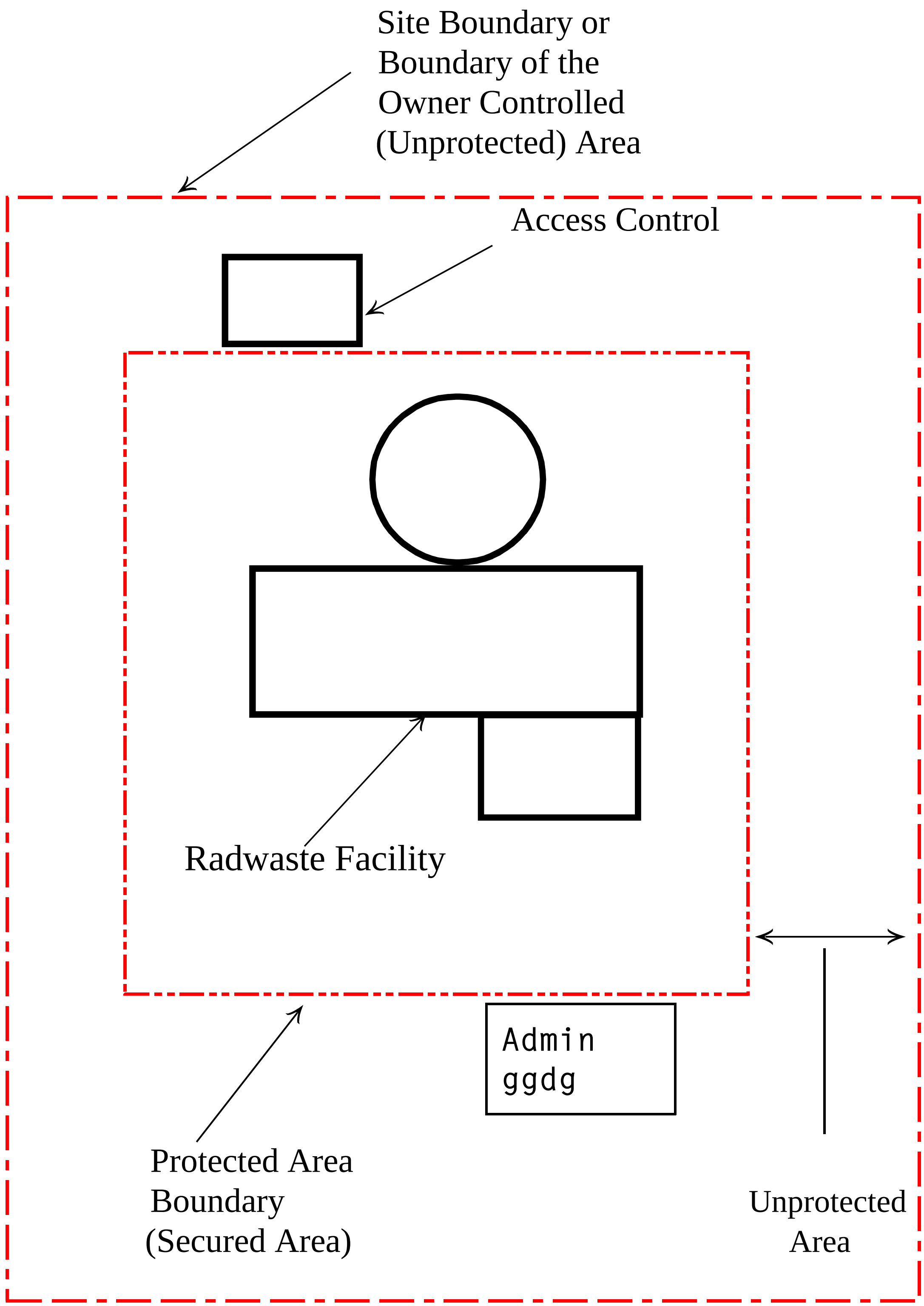

Figure 1 - Informational Schematic Describing
Protected and Unprotected Areas

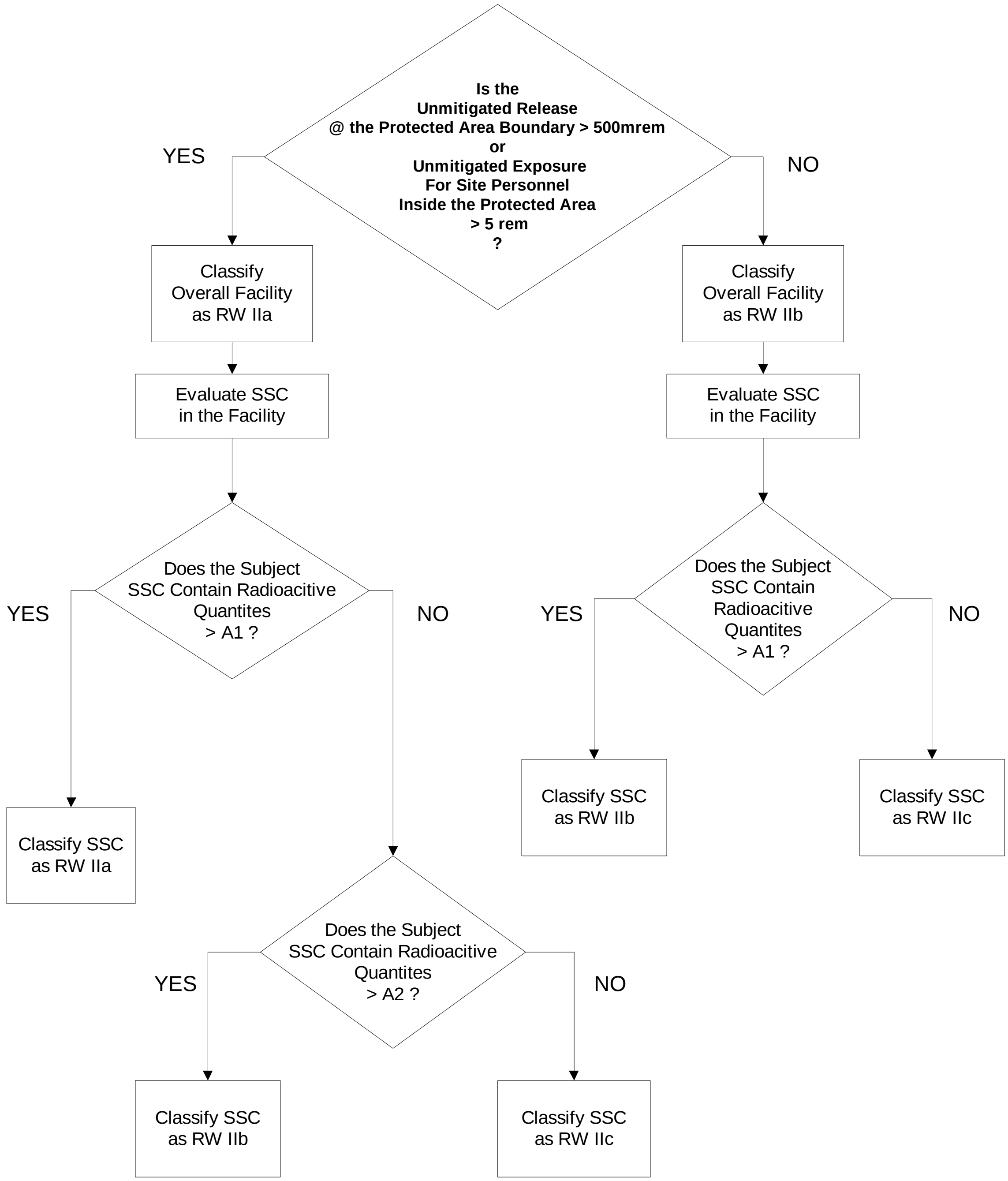

Figure 2 - Flowchart of Safety Classification Process

Appendix A
INDUSTRY CODES AND STANDARDS

American Concrete Institute, ACI-318, "Building Code Requirements for Reinforced Concrete" (ACI 318-89, Revised 1999), 1999.

American Concrete Institute, ACI-349, “Code Requirements for Nuclear Safety Related Concrete Structures," 1997.

American Institute of Steel Construction, N690 (S327), “Nuclear Facilities, Steel Safety-Related Structures For Design and Fabrication,” 1984.

American Institute of Steel Construction, “Specifications for Structural Steel Buildings, Manual of Steel Construction,” 2nd Edition, 1995.

American Institute of Steel Construction, "Specifications for Structural Steel Buildings, Allowable Stress Design and Plastic Design, Manual of Steel Construction,” 9th Edition, 1993.

American Iron and Steel Institute, SG-673, “Specification for the Design of Cold-Formed Steel Structural Members,” August 1986 with December 1989 Addendum.

American Nuclear Society, "Gaseous Radioactive Waste Processing Systems for Light-Water Cooled Reactor Plants," ANSI/ANS-55.4-1993.

American Nuclear Society, "Liquid Radioactive Waste Processing System for Light Water Cooled Reactor Plants," ANSI/ANS 55.6-1993.

American Nuclear Society, "Solid Waste Processing System for Light Water Cooled Reactor Plants," ANSI/ANS-55.1-1992.

American Petroleum Institute, 610, “Centrifugal Pumps for Petroleum, Heavy Duty Chemical, and Gas Industry Services,” 1995.

American Petroleum Institute, 620, “Design and Construction of Large, Welded, Low-Pressure Storage Tanks, 1990.

American Petroleum Institute, 650, “Welded Steel Tanks for Oil Storage,” 1998.

American Petroleum Institute, 674, “Positive Displacement Pumps-Reciprocating,” 1995.

American Petroleum Institute, 675, “Positive Displacement Pumps-Controlled Volume,” 1994.

American Society of Civil Engineers, 7-95, "Minimum Design Loads for Buildings and Other Structures," 1995.

American Society for Mechanical Engineers, Boiler and Pressure Vessel Code, Section II, “Material Specification,” 1999.

American Society for Mechanical Engineers, Boiler and Pressure Vessel Code, Section VIII, “Pressure Vessels,” 1999.

American Society for Mechanical Engineers, Boiler and Pressure Vessel Code, Section IX, “Welding and Brazing Qualification,” 1999.

American Society for Mechanical Engineers, Boiler and Pressure Vessel Code, B31.3, “Process Piping,” 1999.

American Society for Testing & Materials, A36-00, “Standard Specification for Carbon Structural Steel,” 2000.

American Society for Testing & Materials, A500-99, “Standard Specification for Cold-Formed Welded and Seamless Carbon Steel Structural Tubing in Rounds and Shapes,” 1999.

American Society for Testing & Materials, A795-97, “Standard Specification for Black and Hot-Dipped Zinc-Coated Welded and Seamless Steel Pipe for Fire Protection Use,” 1997.

American Society for Testing & Materials, B359-98, “Standard Specification for Copper and Copper-Alloy Seamless Condenser and Heat Exchanger Tubes With Integral Fins,” 1998.

American Society of Mechanical Engineers, QME-1-1997, “Qualification of Active Mechanical Equipment Used in Nuclear Power Plants,” December 31, 1997.

American Welding Society, D1.1, “Structural Welding Code-Steel,” 17th Edition, 2000.

American Welding Society, D1.3, “Structural Welding Code-Sheet Steel,” 1998.

American Welding Society, D9.1, “Sheet Metal Welding Code,” 1990.

American Welding Society, D10.9, “Specification for Qualification of Welding Procedures and Welders for Piping and Tubing,” 1980.

National Electrical Manufacturers Association, Publication Number TC2, “Electrical Polyvinyl Chloride(PVC) Tubing and Conduit,” 1998.

National Electrical Manufacturers Association, Publication Number VE1, “Metal Cable Tray Systems,” 1996.

National Fire Protection Association, NFPA 13, “Installation of Sprinkler Systems,” 1999.

National Fire Protection Association, NFPA 14, “Standard for the Installation of Standpipe Fire Protection, Private Hydrant, and Hose Systems,” 2000.

Sheet Metal and Air Conditioners Contractor National Association, “Seismic Restraint Manual Guideline for Mechanical Systems,” 2nd Edition, 1998.

Tubular Exchanger Manufacturers Association, “Standards of the Tubular Exchanger Manufacturers Association, Eighth Edition,” 2000.

The Codes and Standards are available from:

American Concrete Institute (ACI),

American Institute of Steel Construction (AISC), One E. Wacker Drive, Suite 3100, Chicago, IL 60601-2001.

American Iron and Steel Institute (AISI),1101 17th Street, NW, Washington, DC 20036.

American Nuclear Society (ANS), 555 N. Kensington Avenue, La Grange Park, IL 60525.

American Petroleum Institute (API), 1220 L Street, NW, Washington, DC 20005.

American Society of Mechanical Engineers (ASME), 345 East 47th Street, New York, NY 10017.

American Society for Testing & Materials (ASTM), 100 Barr Harbor Drive, West Conshohocken, PA 19428-2959.

American Welding Society (AWS), 550 NW LeJeune Road, Miami, FL 33126.

National Electrical Manufacturers Association (NEMA), 1300 N. 17th Street, Rosslyn, VA 22209.

National Fire Protection Association (NFPA), Inc., Battery March Park, Quincy, MA 02269.

Sheet Metal and Air Conditioners Contractor National Association (SMACNA), 4201 Lafayette Center Drive, Chantilly, VA 20153-1230.

Tubular Exchanger Manufacturers Association (TEMA), 25 N. Broadway, Tarrytown, NY 10591.

REGULATORY ANALYSIS

1. STATEMENT OF PROBLEM

Revision 1 of Regulatory Guide 1.143, "Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants," was issued in October 1979. This guide furnishes design guidance acceptable to the NRC staff related to seismic and quality group classification and quality assurance provisions for radioactive waste management structures, systems, and components. Further, it describes provisions for controlling releases of liquids containing radioactive materials, e.g., spills or tank overflows, from all plant systems outside reactor containment. Regulatory Guide 1.143 encompassed the design of buildings, structures, systems, and components and referred to several design and construction codes and standards, such as American National Standards Institute (ANSI) N197-1976, ANSI N199-1976, American Nuclear Society (ANS) ANS 55.1-1979, ANS 55.4-1979, American Concrete Institute ACI-318-1977, and American Institute of Steel Construction AISC-1969.

These references are now obsolete or have been superseded by newer ANSI and ANS radioactive waste facility design standards. ANS has recently issued ANS-55.1-92, ANS-55.4-93, and ANS-55.6-93, which are the industry consensus standards currently applicable to the overall design of radioactive waste facilities. In addition, several other referenced codes such as "Building Code and Commentary," ACI-318-77; or "Specification for the Design, Fabrication and Erection of Structural Steel for Buildings," AISC-1969, have been updated and modified since the issuance of Revision 1 of Regulatory Guide 1.143. Also, there has been increased understanding of, and corresponding changes in relation to, radiation exposure and monitoring and quality assurance needs for the design and construction of radioactive waste facilities and the associated systems, structures and components.

The Operating Basis Earthquake (OBE), as was used in Revision 1 of Regulatory Guide 1.143 as the design basis, creates further difficulties. In 1997, the NRC staff revised 10 CFR 100.23 and added Appendix S to 10 CFR Part 50 that essentially state that, if the review level earthquake (OBE) is defined as less than 1/3 of the safe-shutdown earthquake (SSE), no explicit design analysis for the OBE level earthquake will be required. In other words, the revised criteria have effectively eliminated the OBE as a design basis seismic event. In recent staff licensing actions, the Standard (Advanced) Reactor Designs used only a SSE event as the design basis, consistent with the methodology in the recent revision to 10 CFR 100.23 and the addition of Appendix S to 10 CFR Part 50. Thus, Revision 1 of Regulatory Guide 1.143 was almost not usable for standard reactor designs.

The staff maintains that recommendations based on the latest editions of the Design and Construction Standards and Codes mentioned above and reference to current quality assurance standards and NRC regulations provides a means to achieve better evaluation of radioactive waste management systems, structures, and components installed in light water-cooled nuclear power plants.

2. OBJECTIVES

The objective of the regulatory action is to update NRC guidance on the design, construction, and quality assurance of radioactive waste management systems, structures, and components installed in light-water-cooled nuclear power plants.

3. ALTERNATIVES AND CONSEQUENCES OF PROPOSED ACTION

3.1 Alternative 1 - Do Not Revise Regulatory Guide 1.143

Regulatory Guide 1.143 would not be revised and licensees would continue to rely on the current version of Regulatory Guide 1.143 with references from the late 1960s and mid 1970s. The staff acknowledges that many licensees who are presently involved in the design of radioactive waste management systems, structures, and components installed in light-water-cooled nuclear power plants, as a matter of practice, already rely on more recent editions of ANSI and ANS radioactive waste facility design standards and ACI and AISC codes.

3.2 Alternative 2 - Update Regulatory Guide 1.143

The NRC staff has identified the following consequences associated with adopting Alternative 2.

3.2.1 Licensees will use the latest consensus standards available, thereby improving design, evaluation, and quality assurance of radioactive waste management systems, structures, and components. The staff views the latest standards as improved because they incorporate the latest technology and knowledge on the subject.

3.2.2 Regulatory efficiency will be improved by reducing uncertainty as to what is acceptable and by encouraging consistency in the design, evaluation, and quality assurance of radioactive waste management systems, structures, and components. The benefits to both the NRC and industry will be to the extent this occurs. An updated regulatory guide would facilitate NRC review because licensee submittals should be more predictable and consistent analytically. Similarly, licensee's adherence to the latest consensus standards should benefit licensees by reducing the likelihood for follow-up questions and possible revisions to licensees' plans.

3.2.3 An updated regulatory guide could result in cost savings for both the NRC and industry. From the NRC's perspective, relative to the baseline, NRC will incur one-time incremental costs to develop the regulatory guide for public comment and to finalize the regulatory guide. However, the NRC should also realize cost savings associated with the review of licensee submittals. In the staff's view, the continuous and on-going cost savings associated with these reviews should more than off-set this one-time cost.

On balance, it is expected that industry would realize a net savings, as their one-time incremental cost to review and comment on a revised regulatory guide would be more than compensated for by the efficiencies (e.g., reduced follow-up questions and revisions) associated with each licensee submittal.

3.2.4 The use of industry consensus standards that are already being used by licensees would enhance the continued use of the guidance contained in ANS-55.1-92, ANS-55.4-93, and ANS-55.6-93, thereby avoiding costs related to a "new" agency-prepared standard. This approach would also comply with the Commission's directive that standards developed by consensus bodies be utilized per Public Law 104-113, "National Technology and Transfer Act of 1995."

4. CONCLUSION

Based on this regulatory analysis, it is recommended that the NRC revise Regulatory Guide 1.143. The staff concludes that the proposed action will reduce unnecessary burden on both the NRC and its licensees, and it will result in an improved process for the design, evaluation, and quality assurance of radioactive waste management systems, structures, and components. Furthermore, the staff sees no adverse effects associated with a revision to Regulatory Guide 1.143.

BACKFIT ANALYSIS

The regulatory guide does not require a backfit analysis as described in 10 CFR 50.109(c) because it does not impose a new or amended provision in the NRC's rules or a regulatory staff position interpreting the NRC's rules that is either new or different from a previous applicable staff position. In addition, this regulatory guide does not require the modification or addition to systems, structures, components, or design of a facility or the procedures or organization required to design, construct, or operate a facility. Rather, a licensee or applicant may select a preferred method for achieving compliance with a license or the rules or the orders of the Commission as described in 10 CFR 50.109(a)(7). This regulatory guide provides an opportunity to use industry-developed standards, if that is a licensee's or applicant's preferred method.

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ML003740970

SEISMIC SAFETY OF NUCLEAR POWER PLANTS

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Abstract

This paper summarizes the work performed by the International Atomic Energy Agency in the areas of safety reviews and applied research in support of programmes for the assessment and enhancement of seismic safety in Eastern Europe and in particular WWER type nuclear power plants during the past seven years. Three major topics are discussed; engineering safety review services in relation to external events, technical guidelines for the assessment and upgrading of WWER type nuclear power plants, and the Coordinated Research Programme on "Benchmark study for the seismic analysis and testing of WWER type nuclear power plants". These topics are summarized in a way to provide an overview of the past and present safety situation in selected WWER type plants which are all located in Eastern European countries. Main conclusion of the paper is that although there is now a thorough understanding of the seismic safety issues in these operating nuclear power plants, the implementation of seismic upgrades to structures, systems and components are lagging behind, particularly for those cases in which the re-evaluation indicated the necessity to strengthen the safety related structures or install new safety systems.

1. INTRODUCTION

The concern on the safety level of existing nuclear power plants in Eastern Europe came into focus a few years ago. One of the major reasons for this concern was the recognition that some site-related external events were not properly considered in the original plant design. Furthermore, there was need to compare the criteria, standards and methods used to establish seismic safety in eastern European nuclear power plants with those generally accepted in international practice.

Seismic safety issues generally involve two major components; those related to the derivation of the design basis parameters (i.e. seismic input) and those involving the seismic capacity of structures, equipment and distribution systems. Regarding the first component, although most Eastern European nuclear power plant sites can be characterized as low to medium seismicity, the deficiency in the geological and seismological databases as well as the methods used in the 1970s for determining the seismic hazard at a specific site, have led to the necessity to implement comprehensive hazard re-evaluation programmes of those facilities. The results of the new studies consistently indicate that the original design basis ground motion parameters had been underestimated, sometimes by a considerable margin.

The issues related to the seismic capacity of structures, equipment and distribution systems are even more complex. For WWER and RBMK type nuclear power plants, structures which do not function as a pressure boundary are designed like conventional industrial frame buildings, often using precast concrete elements. Moreover, in WWER-440 and RBMK type nuclear power plants, the 'confinement' concept restricts the pressure boundaries to the lower part of the reactor building. WWER-1000 type plants, however, have a proper structural containment and therefore are inherently more robust for external events.

The involvement of the IAEA in the seismic safety issues of Eastern Europe has been substantial through national and regional projects. Seismic safety review missions visited nuclear power plants in Armenia, Bulgaria, Czech Republic, Hungary, Poland, Romania, Russian Federation, Slovakia, Slovenia and Ukraine within the past seven years.

These countries operate different types of nuclear power plants, i.e. WWER-440/230 (Armenia, Bulgaria, Russian Federation, Slovakia), WWER-440/213 (Czech Republic, Hungary, Slovakia, Russian Federation, Ukraine), WWER-1000 (Bulgaria, Czech Republic, Russian Federation, Ukraine), RBMK (Russian Federation, Ukraine), Candu (Romania) and PWR (Slovenia).

The level of IAEA involvement has also varied greatly ranging from minimal in Poland (where the nuclear power programme was abandoned), Russian Federation and Ukraine, to limited in the Czech Republic, Romania and Slovenia, to extensive in Armenia, Bulgaria, Hungary and Slovakia. The extent of the involvement has depended mainly on the urgency of the need as expressed by the host country.

The activities related to the assessment and enhancement of seismic safety may be considered within two time frames. The engineering services, i.e. site/plant specific reviews, are short term actions to provide recommendations to the regulatory authority and the nuclear power plant management regarding criteria and methods of assessment and upgrading. There is also the coordinated research programme dealing with the seismic safety of WWER type plants in the medium and long term. This programme is titled, "Benchmark study for the seismic analysis and testing of WWER type nuclear power plants" and involves 25 institutions from 15 countries. Another coordinated research programme on the "Assessment of RBMK type nuclear power plants in relation to external events" will begin in 1997.

It should also be mentioned that substantial amount of help in terms of supply of equipment, mainly computer hardware and software for seismic hazard and structural analysis, as well as seismic instrumentation, was provided to Eastern European countries under the scope of national technical assistance and co-operation programmes.

These short and long term activities will be described in the subsequent sections of this article with emphasis on the results achieved so far and what remains to be done in order to significantly improve the seismic safety of these nuclear power plants built to earlier standards.

2. REVIEW SERVICES

A seismic re-evaluation programme for a nuclear power plant has three major components, as follows:

(i) the re-assessment of the seismic hazard as an external event;
(ii) the evaluation of the plant specific seismic capacity to withstand the loads generated by such event, and
(iii) the implementation of upgrades to buildings and components, if needed.

Figure 1 shows the general flow diagram for the seismic re-evaluation process, constituted by five major phases, starting with the assessment of the original seismic input and design bases and finishing with the implementation of the upgrades for the structures, systems and components upgrades if required.

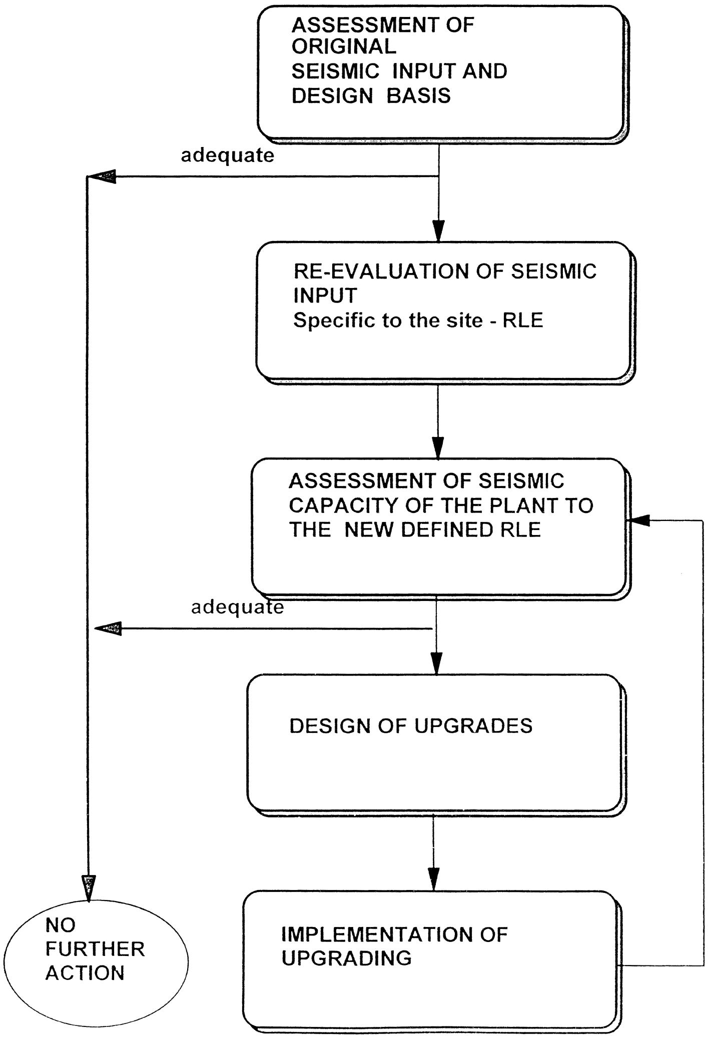

FIG. 1. Flow diagram for seismic re-evaluation and upgrading of existing nuclear power plants.

The IAEA has conducted a substantial number of seismic safety review services to nuclear power plants in 10 East European countries covering 11 sites, the scope of which depended on the stage of assessment and/or upgrading of the specific plant or unit. In most of the cases the process of review started with the assessment of the original seismic input.

The interim results of the re-evaluation of the seismic hazard for Eastern European nuclear power plants are given in Table I.

The geological stability and the ground motion parameters are assessed according to specific site conditions and in compliance with criteria and methods valid for new facilities, which means in accordance with criteria established by the IAEA Safety Guide 50-SG-S I (Rev. 1). Therefore, the review level earthquake RLE should correspond to the SL-2 level directly related to ultimate safety requirements, i.e. a level of extreme ground motion that shall have a very low probability of being exceeded during the plant lifetime and represents the maximum level to be used for design and re-evaluation purposes. As established in the above mentioned IAEA NUSS Safety Guide, the recommended minimum level is a peak ground acceleration of

0.10g for the zero period of the design response spectrum. For the probability of exceedance a typical value of 10^{-4}/yr is usually used coupled with elastic ground response spectra.

Table II provides an overview of the IAEA engineering review services in relation to seismic safety which were conducted to these plants within the past seven years, including a detailed list with all missions, workshops and meetings conducted in that period. Each service is designated with a code indicating the type of review provided in terms of the stage of the assessment (see Figure 1).

TABLE I. SEISMIC SAFETY STATUS OF SELECTED WWER NPPS IN EASTERN EUROPE

Plant	Original SDB	Reassessed SDB (RLE)	Capacity Check	Upgrades to RLE	
				Easy Fixes	Structural
Kozloduy 440	NED	0.2g	Neg.	Yes	No
Kozloduy 1000	0.1 g	0.2g	PSA (*)	No	No
Bohunice V 1	NED	0.25 g?	Neg.	Some	Some
Bohunice V2	NED	0.25g?	Neg.	Some	No
Mochovce	0.06g	0.1 g?	No	No	No
Paks	NED	0.25g	Neg.	Yes	No
Armenia	0.1 /0.2	0.35	No	No	No

Legend:
SDB: **S**eismic **D**esign **B**asis
NED: **N**o **E**xplicit **D**esign
Neg.: Inadequate seismic capacity for the reassessed SDB (RLE)
?: A question mark indicates an ongoing activity with a preliminary indication of the reassessed SD13 (RLE)
No: The activity has not started yet
*: Incomplete.

TABLE II. 5 YEAR SUMMARY OF IAEA SITE/SEISMIC SAFETY REVIEW SERVICES TO EASTERN EUROPEAN NPPS

Country	**Plant**		**Number of services (1990-95)**		
		w	S	SI	SC
Armenia	Armenia	-	-	5	3
Bulgaria	Kozloduy 1-4	1	2	5	5
Bulgaria	Kozloduy 5-6	-	-	1	2
Bulgaria	Belene	-	2	2	-
Croatia	(Site Survey)	-	1	-	-
Czech Republic	Temelin	2	4	-	-
Czech Republic	(Spent Fuel Storage)	-	I	1	-
Hungary	Paks	-	-	6	5
Romania	Cernavoda	1	-	-	2
Russian Federation	(Generic WWER)	1	-	-	-
Russian Federation	Smolensk	-	-	1	1
Slovakia	Bohunice V 1	-	-	-	3
Slovakia	Bohunice V 2	1	-	2	-
Slovakia	Mochovce	1	-	2	3
Slovenia	Krsko	1	-	3	1
Ukraine	Crimea	-	-	-	-
TOTAL		**8**	**10**	**29**	**25**

Legend:
W: Workshop
S: Site Safety Review
SI: Review of Seismic Input and Tectonic Stability
SC: Review of Seismic Capactiy.

Considering that the site related investigations for reassessing the seismic input need a long time for completion (i.e. several years), a conservative preliminary value for the RLE is generally assumed for starting the activities related to the re-evaluation of the seismic capacity and upgrading of plant systems, structures and components. This may be called the interim RLE (iRLE).

Another important consideration for re-evaluation purposes is that if median plus one standard deviation was used for the definition of the peak ground acceleration, a median shaped elastic response spectra as given in US-NUREG/CR-0098, Ref [2], is permitted.

3. CRITERIA FOR RE-EVALUATION OF SEISMIC CAPACITY

In relation to the second component of the programme mentioned in Section 2, the objective is to enhance the seismic safety in compliance with valid standards and recognized practice, using (a) "as-is" data, i.e. data reflecting the present state of the plant items; (b) more realistic criteria and methods than the ones used in the design process for at least those functions, systems, components and structures required to ensure safe shutdown and to maintain it in safe shutdown conditions, trying to avoid unnecessary conservatism. This is often a subset of the structures, systems, and components important to safety. This practice effectively ensures that a set of "dedicated, earthquake-hardened safe shutdown systems" exist at the plant.

Figure 2 provides the flow diagram of the detailed work plan indicating sequence, relationship and interdependence between different tasks. The main steps and criteria used are as follows:

3.1. Identification and classification of seismic safety functions, systems and components

The first step is the identification of the functions, systems, components and structures required during and after an earthquake occurrence. For this purpose, the main criteria and assumptions as indicated by international practice are:

(a) the plant must be capable to be brought to and maintained in a safe shutdown condition during the first 72 hours following the occurrence of the RLE;

(b) safe shutdown means hot or cold shutdown;

(c) simultaneous offsite and plant turbine generated power loss occurs for up to 72 hours;

(d) loss of make-up water capacity from offsite sources occurs for up to 72 hours;

(e) the required safe shutdown systems should fulfil single active failure criterion;

(f) the required safe shutdown systems should include one main path and one redundant path;

(g) other external events such as fires, flooding, tornadoes, sabotage, etc. are not postulated to occur simultaneously;

(h) Loss of Coolant Accident (LOCA) and High Energy Line Breaks (HELB) are not postulated to occur simultaneously.

The safe shutdown equipment list (SSEL) is the list of the minimum set of selected equipment required to achieve and maintain those safe shutdown conditions and is the most important result of this step.

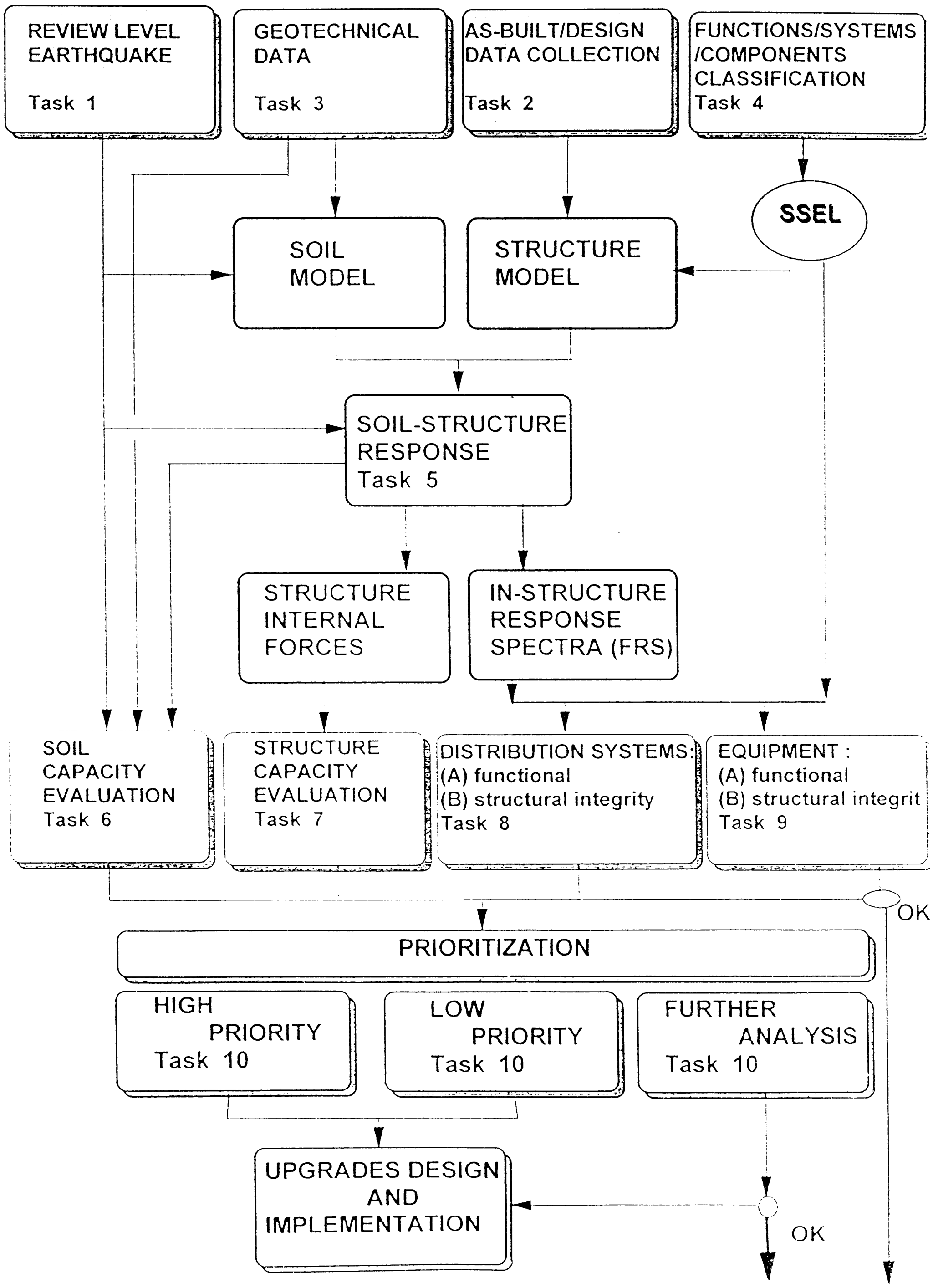

FIG. 2. Detailed flow diagram for the assessment and improvement of seismic safety.

3.2. Plant walkdown

Emphasis should be given to the collection and compilation of original design basis data and documentation in order to minimize the effort required for the re-evaluation programme. In this regard the seismic plant walkdown has become one of the most important components of the seismic re-evaluation programme for an existing facility, with the main objectives of collection of information on as-is conditions and of assessment of the seismic capacity of equipment.

The main focus of the walkdown is on anchorage of the equipment; load path from the anchorage up through the equipment; the equipment structure; and spatial interactions.

In general, there will be three alternative disposition categories for each structure, system and component being evaluated during the walkdown:

(1) Disposition 1: a fix is required;

(2) Disposition 2: the seismic capacity is uncertain and an evaluation is needed to determine if a fix is required, and

(3) Disposition 3: the seismic capacity is adequate for the specified RLE and the items appear to be seismically rugged.

The three alternate dispositions are primarily based on judgement and the walkdown teams must be sufficiently experienced in order to make these judgements.

Screening guidelines are used to determine if the components are represented by the experience database applicable to the component in question. Unfortunately, most of the components and distribution systems in the WWER type rectors were manufactured by organizations for which seismic and testing experience has not yet been gathered and reviewed on an international scale. Similarity analysis should, therefore, be made.

3.3. Evaluation of seismic margin capacity

The concept of High Confidence of Low Probability Failure (HCLPF) capacity is used to assess and quantify the seismic margins of NPPs. In simple terms it corresponds to the seismic input level at which, with high confidence (≥95%) it is unlikely (i.e. ≤5%) that failure of a system, structure or component required for safe shutdown of the plant will occur.

(a) The first step in the estimation of HCLPF seismic capacity is to develop a clear definition of what constitutes failure for each of the systems, structures and components being evaluated. Several modes of seismic failure may have to be considered. It may be possible to identify the failure mode which is most likely or the most dominant to be caused by the seismic event by reviewing the structure, system, component (SSC) design and to consider only that mode.

(b) The response analysis for RLE is conducted with median estimate damping values in accordance with the stress levels. Sufficient parameter variation is considered to account for uncertainties, e.g. soil material properties, and stiffness and mass characteristics of the structures and components. As an example, the damping values recommended for the seismic re-evaluation of the Armenian NPP are indicated in Table IV.

(c) Nearly all structures and components exhibit at least some ductility (i.e., ability to strain beyond the elastic limit) before failure or even significant damage.

The inelastic energy absorption factor, F_{μ} is related to the amount of inelastic deformation that is permissible for each type of structural element. The additional seismic margin due to this inelastic energy absorption ratio (or ductility) should be considered in any margin review. In most cases, it is feasible to use linear elastic analysis techniques.

When linear elastic analysis is applied, the easiest way to account for the inelastic energy absorption capability is to reduce seismic response by the Fμ factor. Fμ is defined as the amount that the elastic-computed seismic demand may exceed the capacity of the component without impairing its performance. It means that for non-brittle (ductile) failure mode inelastic distortion associated with a demand-capacity ratio greater than unity is permissible.

Standard Fμ values for different structural systems as being accepted for WWER type plants are determined considering two conditions: (i) the verification of seismic capacity of existing structures and components at WWER type reactors; and (ii) the verification of seismic capacity of structures designed using joint ductile requirements as established in applicable codes. As an example, the inelastic energy absorption factors recommended for the seismic re-evaluation of the Armenian NPP are indicated in Table V.

TABLE III. PARTITION OF TASKS FOR PARTICIPATING INSTITUTIONS

	Structures		**Components**		**Distribution Systems**	
	Kozloduy NPP	**Paks NPP**	**Kozloduy NPP**	**Paks NPP**	**Kozloduy NPP**	**Paks NPP**
Analysis	IZSIIS (M) Siemens (G) MD (CR) CL (BG)	SAGE (B) IZIIS (M) AEP (RF)- P-NPP (H)- Siemens (G) MD (CR) EQE (BG) EQE (USA) CL (BG)	Siemens (G) VNIIAM (RF) WESE (B) BRI (BG)- SP (CH)	Siemens (G)- P-NPP (H) CKTI (RF)-SA (CR) VNIIAM (RF) Argonne (US)	K-NPP Siemens (G)- WESE (B) SP (CH)-BRI (BG) CL (BG) EQE (US) Wolfel (G)	P-NPP (H)- Siemens (G) CKTI (RF)-SA (CR) WESE (B) EQE (US)
Testing	K-NPP Ismes (I)	Ismes (I) P-NPP (1-1)	IZIIS (M) AEP (RF) VNIIAM (RF) K-NPP	P-NPP (H) VNIIAM (RF) IZIIS (M)	CKTI (RF) VNIIAM (RF)	CKTI (RF) P-NPP (H) VNIIAM (RF)
Experience Data	Siemens (G) SAS (SR)	Siemens (G) SAS (SR)	AEP (RF)- Siemens (G) EQE (USA) VNIIAM (RF) WESE (B) SA (US)	SA (R) EQE (USA) Siemens (G) SA (CR) VNIIAM (RF) WESE (B) SA (US)	SA (R) EQE (USA) AEP (RF) VNIIAM (RF) WESE (B) SA (CR) SA (US)	EQE (USA) AEP (RF) SA (CR) VNIIAM (RF) WESE (B) SA (US)

TABLE IV. DAMPING VALUES TO BE USED FOR SEISMIC RE-EVALUATION OF THE ARMENIAN NPP

ITEMS	DAMPING (% of critical damping)	
	with stress levels < yield	with stress levels ≥ yield
(a) **Structures:**		
(1) Reinforced concrete structures:	7.0 %	10%
(2) Welded steel structures:	5.0 %	7.0%
(3) Bolted or riveted steel structures:	7.0 %	10%
(4) Reinforced masonry walls:	7.0 %	10%
(5) Unreinforced masonry walls:	5.0 %	7.0%
(6) Steel structures with precast panels:	7.0%	7.0%
(b) **Soil:**	For simplified soil-structure interaction analysis (SSI) radiation damping as a function of structural foundation geometry will not be limited but resultant composite modal damping should not exceed in principle, values in typical national standards. [Ref.7]. However, the use of higher values, if properly justified and determined would be permitted.	
(c) **Systems and Components:**	5.0 %	5.0%
except the following:		
(1) Tank liquid sloshing:	0.5%	0.5%
(2) Cable Raceway: if at least one quarter full of loose cable	10.0%	15.0%
(3) HVAC Duct:	7.0%	7.0%
(4) Vertical pumps: (deep well and emersion)	3.0%	3.0%
(5) Instrument racks:	3.0%	3.0%
(d) **Generation of In-structure Spectra:**		
(1) When generating floor in-structure or in component response spectra for relatively lightly loaded supporting structures, systems or components (S S ≤ 0.50 Sy):		
(a) steel:	2.0%	
(b) concrete:	4.0%	
(2) When generating floor, in-structure or in component response spectra for supporting structures (0.5 Sy < S < 1.0 Sy):		
(a) steel:	5.0%	
(b) concrete:	7.0%	
(3)When generating in-structure or in-component response spectra for supporting structure loaded beyond yield (S>= 1.0 Sy):		
(a) steel:	7.0%	
(b) concrete:	10.0%	

TABLE V. INELASTIC ENERGY ABSORPTION FACTORS Fμ (1) TO BE USED FOR SEISMIC RE-EVALUATION OF THE ARMENIAN NPP

Structural System	Fμ (2) (3)
(I)MOMENT RESISTING FRAME SYSTEMS	
***Concrete**:*	
(1) Columns where flexure dominates:	1.25
(2) Columns where axial compression or shear dominates:	1.00 **(4)**
(3) Beams:	1.25
(4) Connections (any):	1.00
Steel:	
(5) Columns where flexure dominates:	1.50
(6) Columns where axial compression or shear dominates:	1.00 **(4)**
(6) Beams:	1.50
(7) Connections (any):	1.00
(II) SHEAR WALLS	
(1) Concrete and Reinforced Masonry Walls:	
(a) in plane bending:	1.75
(b) in plane shear:	1.50
(c) out-of-plane bending:	1.75
(d) out-of-plane shear:	1.00
(2) Unreinforced masonry out-of-plane shear:	1.00
(c)BRACED FRAMES:	
***Concrete**:*	
(1) Columns where flexure dominates:	1.25
(2) Columns where axial compression or shear dominates:	1.00
(3) Beams:	1.50
(4) Bracing (Steel):	1.50
(5) Connections (any):	1.00
Steel:	
(6) Columns:	1.00
(7) Beams:	2.00
(8) Tension only bracing and tension ties or struts:	1.50
(9) Connections (any):	1.00
(d)Adequately Anchored Passive Electrical and Mechanical Equipment:	
(1) Bent plate panels:	1.50
(2) Steel angles framing:	2.00
(3) Steel housings:	2.00
(4) Cast iron:	1.00
(e) Piping, Conduit, Instrument Tubing and HVAC Duct:	
(1) Butt joined grove welded steel pipe:	1.50
(2) Socket welded pipe:	1.50
(3) Threaded pipe:	1.00
(4) Conduit:	1.25
(5) Instrument tubing:	1.50
(6) Cable trays:	1.50
(7) HVAC duct:	1.50
(8) Distribution System Supports:	1.25

Notes to Table V:

(1) The relationship between Fμ and μ is as follows:

Fu = □ if the dominant natural frequency is less than 2 Hz
F□ = (2□-1)'% if the dominant natural frequency is between 2 and 8 Hz
F□ = 1 if the dominant frequency is above 33 Hz
F□ = Transition between (2□ -1)□¹ and 1.0 between 8 and 33 Hz.

(2) These F□ values are recommended for use for seismic re-evaluation purposes of **existing structures, systems and components at "ER type reactors**

(3) The F□ values recommended for connections in structures designed using the improved joint ductility requirements contained in US-ACI-318-92 Chapter 21, for concrete, or the US-SEAOC criteria for structural steel, and semi rigid connections or equivalent may be taken as 1.25.

(4) For components in axial compression with K 1 /r ratio less than 40, F□ may be taken as 1.25.

(5) For metal pressure retaining components if stresses are limited to ASME III - 1992, or earlier code allowables, otherwise F□ = 1.0. The 1995 edition of the Code has higher allowable stresses which in general have not received Regulatory Agency acceptance and in any case shall not be used with Fμ values greater than 1.0.

(d) Seismic response of building structures will be evaluated on the basis of dynamic analysis of models of the soil-structure system. In order to develop appropriate structural models special attention is given to (i) structural configuration and construction details (joints, gaps, restraints and supports); (ii) non structural elements, such as masonry or precast reinforced concrete panels that may modify the structure response. Stiffness and strength of such panels, and those of their attachments to the structure, should be accounted for in the formulation of the models; (iii) as-built material properties and dimensions of structural members; (iv) geotechnical data of foundation materials and their potential implications on the necessity to perform soil-structure interaction analysis, for which direct methods are usually being applied. For soil-structure interaction analysis radiation damping will not be limited but resultant composite modal damping should not exceed in principle values in typical national standards. However, the use of higher values, if properly justified and determined, would be permitted.

(e) Combinations of seismic and non-seismic loads shall be made according to the specific equations (for reinforced concrete structural elements, for masonry walls and precast reinforced concrete panels, component pressure boundaries, supports for piping and pressure components and cable raceways). The reassessed seismic input is defined for each of the horizontal components and the vertical component is assumed as a prescribed ratio of the horizontal input.

(f) The approach recommended may be summarized through the following steps

Step l: calculate elastic seismic demand in members and connections by elastic seismic response analysis, using the elastic response spectrum;

Step 2: calculate the inelastic seismic demand in specific members by dividing the elastic seismic demand from Step 1 by an amount, F‖, representing the inelastic energy absorption factor. F‖ values are provided for various types of structural systems;

Step 3: combine the inelastic seismic demand with the best estimate of concurrent non-seismic demand using unity load factors to determine the total demand;

Step 4: estimate seismic capacity of members and connections by ultimate strength or limit strength provisions in accordance with codes for the appropriate materials (i.e. US-ACI or equivalent national codes for concrete, US-AISC or equivalent national codes for steel), including the appropriate strength reduction factors;

Step 5: evaluate total demand to capacity ratios for members and connections from the results of Steps 3 and 4. The structural system and individual members and connections must comply with the structural evaluation criteria when that ratios are less than unity. When those ratios values exceed unity significantly, strengthening measures should be considered.

(g) An earthquake experience and test based judgmental procedure to verify the seismic adequacy of the specified safety-related equipment in operating NPPs using seismic experience methods, was developed in the USA to address regulatory requirements for requalification of older plants. The procedure is primarily based upon the performance of installed mechanical and electrical equipment in conventional plants or other industrial facilities which have been subjected to actual strong motion earthquakes as well as upon the behaviour of equipment components during simulated seismic tests. With a number of caveats and exclusions for excitations below spectra normalized to 0.30g and in some cases 0.50g, for the zero period ground acceleration (i.e. ZPGA), it is unnecessary to perform explicit seismic analysis or test qualification of existing equipment to demonstrate functionality after the strong shaking has ended. The existing database reasonably demonstrates the seismic ruggedness of existing equipment up to these seismic motion bounds. This conclusion should not be extrapolated beyond the classes of equipment existing in the database.

(h) The issue of adequate anchorage is perhaps the most important single item which affects the seismic performance of distribution systems and components, which can slide, overturn, or move excessively when not properly anchored. Adequate strength of system and component anchorage can be determined by any one of many commonly accepted methods. The load or demand on the anchorage system can be obtained from the floor response spectral acceleration for the prescribed damping value and at the estimated fundamental or dominant frequency of the system or component. A conservative estimate of the spectral acceleration may be taken as the peak of the applicable spectra. This acceleration is then applied to the mass of component or system at its center of gravity.

Generally, the four main steps for evaluating the seismic adequacy of equipment anchorage include: anchorage installation inspection; anchorage capacity determination; seismic demand determination; and comparison of capacity to demand.

(i) In addition to the inertia effects there may also be significant secondary stresses induced in systems and components by differential or relative anchor motion if the system or component is supported or restrained at two or more points. For supports it is common practice to evaluate such seismic induced anchor motion, where the relative or differential motion of the building structure at the different points of attachment should be input to a model of the multiple supported component or system. Resultant forces, moments and stresses in the support system determined from the seismic anchor motion effects acting alone shall meet the same limits for normal operation plus RLE inertia stresses.

4. CO-ORDINATED RESEARCH PROGRAMMES

4.1. Background

A coordinated research programme on the benchmark study for the seismic analysis and testing of WWER type nuclear power plants was initiated subsequent to the request from representatives of member states during the IAEA Technical Committee Meeting on the seismic safety of existing nuclear power plants held in Tokyo in August 1991. The conclusions of this meeting called for the harmonization of methods and criteria used in member states in issues related to seismic safety. In particular, seismic safety concerns related to WWER type nuclear power plants were expressed.

With this objective in mind, it was decided that a benchmark study is the most effective way of achieving the principal objective. Two types of ex-USSR designed WWER reactors (WWER-1000 and WWER-440/213) were selected for the benchmark exercise.

Twenty five internationally recognized institutions (public or private companies) from 15 countries take part in the seismic analysis and/or testing of the two prototypes which have been identified as Kozloduy NPP Unit 5/6 and Paks NPP, representing the WWER-1000 and WWER-440/213 respectively.

Four research coordination meetings were held so far, in Paks, Kozloduy, St. Petersburg and Bergamo. Reconnaissance plant walkdowns were performed during the first two meetings for the two selected prototypes.

Thirteen volumes of research material has been prepared by the participating institutions. One of the major activities of the program has been the full scale dynamic testing of the Paks and Kozloduy NPPs using blast excitation.

4.2. Prototype plants

Paks NPP

Paks NPP comprises four WWER-440/213 units. It is located about 100 kms south of Budapest on the Danube river. In the original design of the plant seismic loads had not been taken into consideration. The seismic input for the plant has been recently re-assessed to be 0.25g having site specific response spectra. A major program of seismic evaluation and upgrading is underway at Paks NPP. The so called "easy fixes" have already been implemented. These mainly include equipment supports and anchorages, as well as strengthening of unreinforced masonry walls with the potential of collapsing on safety related items.

Structurally, the WWER-440/213 type NPPs lack a containment, i.e. protection from external loads. The reactor building structure of Paks NPP is steel frame with infill walls and without proper bracing to resist lateral loads. The monolithic concrete part of the building is in the lower part of the structure and serves as an ultimate pressure boundary for extreme internal loads.

Kozloduy NPP Unit 5/6

Kozloduy NPP site has four WWER-440/230 units and two WWER-1000 units. Units 5 and 6 refer to the 1000 MW(e) units. The site is located north from Sofia and on the right bank of the Danube. The soils can be classified as medium with pockets of looser sands especially under parts of the water intake canals. Originally Units 5 and 6 were designed to 0.10g. The reassessed seismic design level is 0.20g associated with a wide band response spectrum rich in lower frequencies (mainly due to the Vrancea earthquake source). Although considerable work has been done in terms of re-evaluation and upgrading of the `easy fixes' type for the smaller units at Kozloduy (these units were not designed for seismic loads originally), so far only a partially completed seismic PSA was performed for Unit 5.

Structurally, WWER-1000 units are radically different from the WWER-440 units. The containment structure of the reactor building provides general protection from extreme external hazards. However the adequacy of this protection with respect to site seismicity still needs consideration.

4.3. Participation and tasks

In the fourth year of its implementation, the number of participating institutions to the co-ordinated research program has increased to 25, coming from 15 member states. Each participating institution (generally a public or private company) has a well defined work plan and task. The distribution of tasks is generally made during the research coordination meetings.

The areas of interest are grouped in a matrix form and may be related to analysis, testing or experience data pertaining to structures, equipment or distribution systems. The application could be either for the Kozloduy NPP Unit 5/6 (i.e. WWER-1000) or the Paks NPP (i.e. WWER-440/213). Each participating institution identifies the area(s) of interest for the coming year during the research coordination meeting. A typical matrix showing the partition of tasks is given in Table III.

After determining the area(s) of interest of the institutions, a work plan is prepared in terms of concrete tasks, identifying the scope of the task, participating institutions in the performance of the task, coordinator of the task and the date of completion of the task. The following is the summary work plan (titles only) which was prepared in June 1996.

Task 1. Safe shutdown systems identification/classification (task completed)

Task 2. Design regulations, acceptance criteria, loading combinations (task completed)

Task 3. Seismic input, soil conditions (task completed)

Task 4. Standards, criteria - comparative study (task continuing)

Task 5. Walkdown of reference plants (Paks and Kozloduy Unit 5 (task completed)

Task 6a. Dynamic analysis of Kozloduy NPP Unit 5 Reactor Building for seismic input (task completed)

Task 6b. Dynamic analysis of Paks NPP Reactor Building for seismic input (task completed)

Task 7. Dynamic analysis of Paks NPP structures (benchmarking with results of Task 8)

Task 7a. Reactor building (task continuing)

Task 7b. Stack (task continuing)

Task 7c. Worm tank (task continuing)

Task 8a. Full scale blast testing of Paks NPP (task completed)

Task 8b. Full scale blast testing of Kozloduy NPP Unit 5 (task completed)

Task 9. Shaking table experiment for selected components (task continuing)

Task 10. On site testing of equipment at Paks and Kozloduy NPPs (task completed)

Task 11. Previous component data (task continuing)

Task 12. Experience data from Vrancea and Armenia earthquakes (task continuing)

Task 13. Experience data from US earthquakes (task completed)

Task 14. Special topic 1 - Assessment of containment dome prestressing of Kozloduy NPP (task continuing)

Task 15. Special topic 2 - Assessment of containment dome/cylindrical part for different loading combinations (task continuing)

Task 16. Special topic 3 - Stress analysis of safety related piping of Kozloduy NPP (task continuing)

Task 17. Special topic 4 - Dynamic analysis of selected structures of Kozloduy NPP (task continuing)

Task 18. Paks NPP feedwater line analysis to be compared with testing which was already performed (task continuing)

Task 19. Analysis of buried pipelines for KNPP [between DG and spray pools] (task continuing)

Task 20. Analysis of buried pipelines for PNPP (task continuing)

Task 21. Comparison of beam vs 3D models for KNPP and PNPP structures (task continuing)

Task 22. Experience data base (WWER SQUG) initiation (task continuing)

Task 23. Consolidation of results and reports (task continuing)

Task 24. Dynamic analysis of Kozloduy NPP Unit 5 structures [benchmarking with results of Task 8] (task continuing)

Task 25. Comparison of blast and vibrator tests for KNPP (task continuing)

Thirteen volumes of research material has been compiled reflecting the results of the completed tasks. These volumes are titled as follows:

Volume 1.	Data related to sites and plants (Pales and Kozloduy NPPs)
Volume 2.	Generic material: codes, standards, criteria
Volumes 3A, 313, 3C, 3D, 3E.	Kozloduy NPP, Units 5/6: Analysis/testing

Volumes 4A, 413, 4C, 4D. Paks NPP: Analysis/testing

Volumes 4A, 5B. Experience data

4.4. Full scale dynamic test of Paks and Kozloduy NPPs

One of the most significant tasks already completed is the full scale dynamic testing of the Paks NPP. The test was conducted by Ismes, an Italian consulting company and Paks NPP with assistance from local contractors especially for the realization of the blasts. The test was performed in December 1994 following a two week preparation period for placing the instruments and recording of smaller test blasts.

The blast location was about 2.5 kilometers from the reactor building. Two main blasts were performed with a total each of 300 kilograms of TNT charge. Three free field locations were selected for instrumentation. Two of these had two borehole (at 40 meters and 15 meters depth) and one surface recording. About 40 meters corresponds to the depth of the firmer geological formation. An additional (fourth) instrument was located about 12 kilometers away in order to provide some information on attenuation characteristics. A large number of seismometers and accelerometers were mounted in the reactor building (some also in other buildings) to record the structural response. Instruments were also placed on certain heavy components and tanks.

Both blasts used a time delay to enhance the duration of the motion so that an adequate time series analysis was possible. In most locations a motion of about 20 seconds was recorded. The records are of very high quality. It should also be noted that all of the instruments functioned as intended.

One set of free field recordings have been made available to the benchmark programme participants. Locations and directions of the in-structure instruments have been indicated and the participants have been asked to make a blind prediction of the response recorded at these locations. All the relevant dynamic soil properties and structural properties have been provided to the participants.

A similar test was carried out for the Kozloduy NPP Unit 5 in June 1996. The test was again performed by Ismes and Kozloduy NPP. Local contractors also participated in the test. All instruments, both free field and in-structure, functioned as intended. The results of the test have been recently processed.

5. CONCLUDING REMARKS

A review and comparison of Figure 1 and Table I, presented earlier reveal some indication of present picture of the seismic safety situation of nuclear power plants with which the IAEA had significant involvement.

The first observation from Table I is that the reassessment of the seismic design basis has been completed for three of the sites (i.e. Kozloduy, Paks and Armenian NPPs) while for Bohunice and Mochovce NPP sites this activity is continuing. For all the sites in question, the reassessment has yielded significantly greater RLE values. This, in turn, indicates that for most of the plants, the capacity check yields the result that the plant requires upgrading (i.e. inadequate seismic capacity).

The last two columns of Table I generally indicates good progress in easy fixes, i.e. mainly supports and anchorages of mechanical and electrical components. For some cases, this included more substantial upgrades involving replacement of batteries and strengthening of unreinforced masonry walls to prevent spatial interaction. Similar progress is unfortunately not the case for structural upgrades or when the installation of additional safety systems were required. Due to bigger funding and longer outage requirements, structural upgrades will probably take much longer to complete. Unfortunately, the overall seismic safety of these NPPs will not have been improved to the target levels, until structural upgrades are implemented.

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In particular, the criteria for the re-evaluation and upgrading of existing NPPs were developed by several authors led by Mr. J. D. Stevenson.

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ANNEX 1

ENGINEERING SAFETY ADVISORY SERVICES Related to Site and External Hazards					
Year 1989					
NO.	TYPE	COUNTRY	**NPP/LOCATION**	DATE	PLANT TYPE
1.	S	Iraq	Site Survey	February 89	-.-
2.	S	Tunisia	Site Survey	April 89	-.-
3.	S	Indonesia	Muria Peninsula	May 89	(not defined yet)
4.	S	USSR	Gorki DHP	June 89	District heating
5.	S	Morocco	Sidi Boulbra	December 89	-.-
Year 1990					
6.	S	Poland	Zarnowiecz	March 90	WWER-440/213
7.	S	CSFR.	Temelin	April90	WWER-1000
8.	S	Iraq	Near Tikrit	May 90	-.-
9.	S	Bulgaria	Belene	June 90	WWER-1000
10.	S	Bulgaria	Kozloduy	June 90	WWER-440/230-1000
1 1.	SC	Romania	Cernavoda	September 90	PHWR 600
12.	S	Pakistan	Chashma	November 90	PI HWR 300
13	SC	Romania	Cernavoda	December 90	PI HWR 600
Year 1991					
14.	S	Indonesia	Muria Peninsula	January 91	(not defined vet)
15.	S	Slovenia	Krsko	March 91	PWR 600
16.	SC	Bulgaria	Kozloduy	April 91	WWER-440/230
17.	W	Bulgaria	Kozloduy	May 91	WWER-440/230
18.	S	Tunisia	NPP Site Survey	May 91	-.-
19.	SI	USSR	Crimea	June 91	WWER-1000
20.	SI-F	Bulgaria	Kozloduy	July 91	WWER-440/230-1000
21.	W	Romania	Cernavoda	September 91	PHWR 600
22.	W	CSFR	Temelin	September 91	WWER-1000
23.	SC	CSFR	Bohunice	September 91	W WER-440/230
24.	SI-F	Bulgaria Kozloduy		November 91	WWER-440/230-1000
25.	S	Tunisia	Site Survey	December 91	--
26.	WP	Indonesia	Muria Peninsula	December 91	(not defined yet)
27.	WP	CSFR	Temelin	December 91	WWER-1000

S: Review of site investigations for all disciplines involved.
S-F: Follow-up mission of previous reviews of site investigations.
SI: Review of investigations for determining the seismic input parameters, specific to the site
SI-F: Follow-up mission of previous reviews of'seismic input definition.
SC: Review of seismic capacity and necessary upgrading of systems, structures and components (SSC) of the plant.
SC-F: Follow-up mission of previous reviews ofseismic capacity and upgrades of SSC.
W: Workshop.
W P: Review of work plans and technical procedures for the site and seismic salety assessment.
B: Activities related to benchmark project for seismic safety of WWER type NPPs.

ENGINEERING SAFETY ADVISORY SERVICES Related to Site and External Hazards					
Year 1992					
NO.	**TYPE**	**COUNTRY**	**NPP/LOCATION**	**DATE**	**PLANT TYPE**
28.	SI	Bulgaria	Kozloduy	February 92	W WER-440/230
29.	W-WP	Slovenia	Krsko	March 92	PWR 600
30.	SI	Bulgaria	Kozloduy	April92	WWER-440/230
31.	SC-F	CSFR	Bohunice	May 92	WWER-440/230
32.	SI-SC	Armenia	Medzamor	May 92	WWER-440/230
33.	S-F	CSFR	Temelin	June 92	WWER-1000
34.	W-S	Malaysia	Site Survey	June 92	- -
35.	SC	Bulgaria	Kozloduy	August 92	WWER-440/230
36.	SC	Pakistan	Chashma	August 92	PWR 300
37.	S	Indonesia	Muria Peninsula	September 92	(not defined yet)
38.	SI	Slovenia	Krsko	October 92	PWR 600
39.	S-F	Indonesia	Muria Peninsula	November 92	(not defined vet)
40.	SC	Bulgaria	Kozloduy	November 92	W WER-400/230
41.	S	Tunisia	Site Survey	December 92	-.-
Year1993					
42.	S-WP	Indonesia	Muria Peninsula	February 93	(not defined yet)
43.	SI-F	Bulgaria	Kozloduy	February 93	WWER-1000, 440/230
44.	SC	Pakistan	Chashma	March 93	PWR 300
45.	S-1	Czech Republic	Temelin	April 93	WWER-1000
46.	SC-F	Slovakia	Bohunice	April 93	WWER-440/230
47.	S-WP	Indonesia	Muria Peninsula	April 93	(not defined yet)
48.	W	Pakistan	Chashma	May 93	PWR 300
49.	SC	Pakistan	Kanupp	May 93	PHWR
50.	S	Croatia	Site Survey	June 93	-.-
51.	SC	Russian Federation	Smolensk	June 93	RBMK
52.	W	China	(Generic)	July 93	-.-
53.	S-F	Indonesia	Muria Peninsula	July 93	(not defined yet)
54.	B-W	I-lungary	Paks	September 93	WWER-440/213
55.	WP	Armenia	Medzamor	August 93	WWER-440/230
56.	SI	Bulgaria	Belene	September 93	WWER-1000
57.	SI	Slovakia	Bohunice	October 93	WWER-4401'230-213
58.	SI	Slovakia	Mochovce	October 93	WWER-440-213
59.	SI-WP	Armenia	Medzamor	November 93	WWER-440/230
60.	S	Indonesia	Muria Peninsula	November 93	(not defined yet)
61.	S	Morocco	Sidi Boulbra	November 93	-.-
62.	SC	Hungary	Paks	December 93	WWER-440/213
63.	SC-F	Pakistan	Chashma	December 93	PWR 300
64.	W	Turkey	Akkuvu	December 93	(not defined yet)

ENGINEERING SAFETY ADVISORY SERVICES Related to Site and External Hazards Year1994					
NO.	**TYPE**	**COUNTRY**	**NPP/LOCATION**	**DATE**	**PLANT TYPE**
65.	*S-F*	Indonesia	Muria Peninsula	February 94	not defined vet)
66.	SI-SC	Bulgaria	Kozloduv	March 94	WWER-1000
67.	SC	Bulgaria	Kozlodu	March 94	WWER-440/230
68.	W	Slovakia	Bohunice	March 94	WWER-440/230
69.	SC	Hungary	Paks	March 94	WWER-440/213
70.	W	Ar entina	(Generic)	April 94	--
71.	B-W	Bulgaria	Kozloduv	June 94	WWER-1000
72.	*S-F*	Czech Republic	Temelin	June 94	WWER-1000
73.	SI	Armenia	Medzamor	Julv 94	WWER-440/230
74.	SC	Slovakia	Mochovce	July 94	WWER-440/213
75.	*SC-F*	Hun ary	Paks	Julv 94	WWER-440/213
76.	S	Indonesia	Muria Peninsula	August 94	(not defined vet)
77.	WP	Slovakia	Mochovce	September 94	WWER-440/213
78.	B	Hun ary	Paks	September 94	W WER-440/213
79.	SC	Armenia	Medzamor	September 94	W WER-440/213
80.	*SC-F*	Bulgaria	Kozloduv	October 94	WWER-440/230
81.	S	Bulgaria	Belene	October 94	W WER-1000
82.	S	Bulgaria	Kozlodu	October 94	WWER-1000/440-230
83.	W	Korea	(Generic)	October 94	- -
84.	SI-F	Slovakia	Mochovce	November 94	WWER-440/213
85.	*SI-F*	Slovakia	Bohunice	November 94	WWER-440/230-213
86.	SC	Armenia	Medzamor	November 94	WWER-440/230
87.	B	Hun -	Paks	December 94	WWER-440/213

S: Review, of site investigations for all disciplines involved.
S-F: Follow-up mission of previous reviews of site investigations.
SI: Review of investigations for determining the seismic input parameters, specific to the site
SI-F: Follow-up mission of previous reviews of seismic input definition.
SC: Review of seismic capacity and necessary upgrading of systems. structures and components (SSC) of the plant.
SC-F: Follow-up mission of previous reviews of seismic capacity and upgrades of SSC.
W: Workshop.
WP: Review of work plans and technical procedures for the site and seismic safety assessment.
B: Activities related to benchmark project for seismic safety of WWER type NPPs.

ENGINEERING SAFETY ADVISORY SERVICES Related to Site and External Hazards Year 1995					
NO.	**TYPE**	**COUNTRY**	**NPP/LOCATION**	**DATE**	**PLANT TYPE**
88.	SC-F	Pakistan	Chashma	Janu 95	PWR 300
89.	SI-F	Hun ary	Paks	Janua 95	WWER-440/213
90.	S-F	Indonesia	Muria Peninsula	March 95	not defined vet
91.	SC-F	Bulgaria	Kozlodu -5	March 95	WWER-1000
92.	SC-F	Slovakia	Mochovce	April 95	WWER-440/213
93.	SI-F	Hungary	Paks	April 95	WWER-440/213
94.	SI-F	Armenia	Medzamor	April 95	WWER-440/230
95.	S	Czech Rep.	(not defined vet	Ma 95	Se pent Fuel Storage
96.	S	Thailand	not defined et	May 95	not defined yet)
97.	SI-F	Armenia	Medzamor	May 95	WWER-440/230
98.	SI/SC-F	Kazakhstan	Alma Ata	May 95	WWER-10 Res. Reactor
99.	SI/SC	Uzbekistan	Tashkent	May 95	WWER-10 Res. Reactor
100.	SI-F	Hungary	Paks	June 95	i WWER-440/213
101.	B-W	Russia	(not applicable)	June 95	W'WER type reactor
102.	S-F	Indonesia	Muria Peninsula	July 95	(not defined yet)
103.	S	Thailand	site selection process)	July 95	(not defined yet)
104.	SI-F	Bulgaria	Belene	July 95	WWER-1000
105.	S-F	Morocco	Sidi Boulbra	September 95	WWER-1000
106.	SC-F	Pakistan	Chashma	September 95	PWR 300
107.	S-F	Indonesia	Muria Peninsula	November 95	(not defined yet)
108.	S	Thailand	(site selection process)	November 95	not defined vet
109.	SI	Iran	Bushehr	December 95	PWR 13-WWER-1000
110.	SI-F	Hun a	Paks	November 95	W WER-440/213
111.	W	Korea	Regional	December 95	Generic
112.	SI-F	Czech Rep.	Skalka	December 95	Sent Fuel Storage

S: Review of site investigations for all disciplines involved.
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SI-F: Follow-up mission of previous reviews of seismic input definition.
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SC-F: Follow-up mission of previous reviews of seismic capacity and upgrades of SSC.
W: Workshop.
WP: Review of work plans and technical procedures for the site and seismic safety assessment.
B: Activities related to benchmark project for seismic safety of W WER type NPPs.

ENGINEERING SAFETY ADVISORY SERVICES Related to Site and External Hazards Year 1996					
NO.	**TYPE**	**COUNTRY**	**NPP/LOCATION**	**DATE**	**PLANT TYPE**
113.	WP	Armenia	Medzamor	January 96	WWER-440/230
114.	SI-F	Hungary	Paks	January 96	WWER-440/213
115.	WP	Slovakia	Bohunice	January 96	WWER-440/213
116.	SI-F	Slovenia	Krsko	February 96	PWR 600
117.	SC	Slovenia	Krsko	February 96	PWR-600
118.	W	Armenia	Medzamor	March 96	WWER-440/230
1 19.	S-F	Indonesia	Muria Peninsula	April 96	(not defined yet)
120.	B	Bulgaria	Kozloduy	June 96	WWER-1000
121.	B-W	Italy	(not applicable)	June 96	W WER type reactors
122.	SC-F	Pakistan	Chashma	June 96	PWR 300
123.	SC-F	Armenia	Medzamor	July 96	WWER-440/230
124.	SC	Armenia	Medzamor	September 96	WWER-440/230
125.	W	Korea	Generic	September 96	-
126.	SC-F	Armenia	Medzamor	November 96	W WER-440/230
127.	S-F	Indonesia	Muria Peninsula	December 96	(not defined yet)

S: Review of site investigations for all disciplines involved.
S-F: Follow-up mission of previous reviews of site investigations.
SI: Review of investigations for determining the seismic input parameters, specific to the site
SI-F: Follow-up mission of previous reviews of seismic input definition.
SC: Review of seismic capacity and necessary upgrading of systems, structures and components (SSC) of the plant.
SC-F: Follow-up mission of previous reviews of seismic capacity and upgrades of SSC.
W: Workshop.
WP: Review of work plans and technical procedures for the site and seismic safety assessment.
B: Activities related to benchmark project for seismic safety of WWER type NPPs.

ENGINEERING SAFETY ADVISORY SERVICES RELATED TO SITE AND EXTERNAL HAZARDS (ESRS)

YEAR	TOTAL NUMBER OF MISSIONS
1989	5
1990	8
1991	14
1992	14
1993	23I
1994	23
1995	25
1996	15
TOTAL	127

NUMBER OF EXTERNAL EXPERTS PER YEAR: 76
(AVERAGE OF LAST TWO YEARS)

IAEA RESOURCES:
1 STAFF MEMBER
1 STAFF MEMBER AS TA

ANNEX 2

DRAFT WORKPLAN
1996/97

Task 1. **Safe shutdown systems identification/classification**
Paks NPP (WWER-440/213) - **Task completed**
Kozloduy NPP (WWER-1000) - **Task completed**
(Co-ordinated by WESE)

Task 2. **Design regulations, acceptance criteria, loading combinations**
AEP will provide FRS for Kozloduy NPP. Otherwise **task completed.**

Task 3. **Seismic input, soil conditions**
Task completed for Kozloduy NPP.
Final soil parameters and seismic input will be sent by Paks NPP **(June 1996)**

Task 4. **Standards, criteria - comparative study**
First phase completed. For the second phase comparison of re-evaluation criteria will be made with original Soviet rules at the time of design. **(October 1997)**

SA (CR) - coordinator
MD (CR)
CKTI (RF)

Task 5. **Walkdown of reference plants (Paks and Kozloduy Unit 5)**
Task completed for both plants.

Task 6a. **Dynamic analysis of Kozloduy NPP Unit 5 RB for seismic input**

AEP (RF) - Siemens - Coordinator
EQE (BG)
MD(CR)
CL(BG)

Task completed.

Task 6b. **Dynamic analysis of Paks NPP RB for seismic input**

AEP (RF) - Siemens (G) - coordinator MD (CR)

Task completed.

Task 7. **Dynamic analysis of Paks NPP structures (benchmarking with results of Task 8)**

Task 7a - Reactor building

Participants: Siemens (G)
EQE (BG)
CL (BG)
MD (CR)
IVO (F)

Referee: Ismes (I)

Input: Distributed by IAEA.

Soil Data: Revised soil data to be distributed by Paks NPP (**June 1996**)

Output: Indicated points on the basemat (4-6, 14-16, 21, 33), at elevation +18.9 (35-37) and on the steel structure (46-47).

Response parameter: acceleration time histories

Format: on diskette or b;, e-mail to Mr. Zola (Ismes) plus a hard copy

Transmittal of "response" by participants (September 1996)

Comparative report by Ismes (December 1996)

Task 7b. Stack

Participants: SAGE (B)
IZIIS (M)

IZIIS (M) will prepare final report. **(October 1996)**

Task 7c. Worm tank

Participants:
AES (US) (in cooperation with Japanese institutes)
PNPP (H)
SA (CR)

AES (US) will evaluate results of experiments conducted so far**. (October 1997)**
PNPP (H) will process blast results and compare with shaking table test results. **(October 1997)**
SA (CR) will study seismic behavior of the tank for sliding. **(October 1997)**

Task 8a. Full scale blast testing of Paks NPP

Participants: PNPP (H) and Ismes (I)

Task completed.

Task 8b. Full scale blast testing of Kozloduy NPP Unit 5

Participants: KNPP (BG), CL (BG), Ismes (1) **(July 1996)**

Task 9. Shaking table experiment for selected components
Participants: IZIIS (M) - coordinator, KNPP (BG), PNPP (H)

Five different types of relays will be tested each from Kozloduy and Paks NPPs.

Delivery of relays: **(September 1996)**
Testing: **(December 1996)**

Task 10. On site testing of equipment at Paks and Kozloduy NPPs

Participants: VNIIAM (RF) - coordinator, PNPP (H), KNPP (BG)

Task continuing.

Task 11. Previous component test data

Participants: IZIIS (M), EQE (BG), KNPP (BG), PNPP (H), AEP (RF), VNIIAM (R-F), CKTI (RF), SA®

SA(R) will compile a list using the information in the Working Material provided by the other participants as well as Eurotest. (December 1996)

Task 12. Experience data from Vrancea and Armenia earthquakes

Participants: AEP (RF), SA (R), EQE (US)

Task continuing for Vrancea data. **(December 1996)**

Task 13. Experience data from US earthquakes

Participants: EQE (US), WESE (B), SA (US)

Task continuing. EQE (US) and SA (US) will meet and discuss in two weeks.

Task 14. Special Topic 1 - Assessment of containment dome prestressing for KNPP

Participants: KNPP (BG), SP (CH), BRI (BG)

New tendons will be designed using Swiss technology, material for 10 tendons for the cylindrical part of the RB will be delivered, a new monitoring system will be evaluated and implemented. **(October 1997)**

Task 15. Special Topic 2 - Assessment of containment dome/cylindrical part for different loading combinations

Participants: KNPP (BG), SP (CH), BRI (BG), EQE (BG)

Task continuing. **(June 1997)**

Task 16. Special Topic 3 - Stress analysis of safety related piping for KNPP

Participants: SP (CH) - co-ordinator, KNPP (BG), Woelfel (G), CKTI (RF)

Task for Woelfel completed. SP (CH) will perform the following: seismic capacity evaluation of remaining piping systems, recommendations for upgrade measures, redesign of support structures where upgrades are necessary, and implement new support structures. **(October 1997)**

Task 17. Special Topic 4 - Dynamic analysis of selected structures of KNPP

Participants: SP (CH) - coordinator, KNPP (BG), BRI (BG), EQE (BG)

Diesel generator building analysis finished by BRI, interaction with underground reservoirs is in progress. (October 1997)

EQE (BG) submitted report on stack to KNPP who will transmit to IAEA.

Other stack analysis finalized by SP (CH).

Task 18. Paks NPP feedwater line analysis to be compared with testing which was already performed

Participants: PNPP (H), CKTI (RF), SA (CR), WESE (B)

Task continuing. **(December 1996)**

Task 19. Analysis of buried pipelines for KNPP (between DG and spray pools)

Participants: EQE (US), Siemens (G)

Task continuing. **(October 1997)**

Task 20. Analysis of buried pipelines for PNPP

Participants: SAGE (B), Siemens (G), SA (CR)

Task continuing. **(October 1997)**

Task 21. Comparison of beam vs 3D models for KNPP and PNPP structures

Participants: MD (CR), EQE (BG)

Task continuing. **(October 1997)**

Task 22. Experience data base (WWER SQUG) initiation

Participants: PNPP (H), KNPP (BG), EQE (US), SA (US), SA (R), SA (CR)

A format will be prepared by SA ® and SA (CR). KNPP and PNPP will check feasibility of providing a sample for database. (December 1996)

LIST OF PARTICIPANTS

Asmis, K.	Safety Evaluation Division (Engineering), Directorate of Analysis and Assessment, Atomic Energy Control Board, Ottawa, Canada
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