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NUCLEAR ENERGY AGENCY
COMMITTEE ON THE SAFETY OF NUCLEAR INSTALLATIONS

Workshop on the Seismic Re-evaluation of all Nuclear Facilities

Workshop Proceedings

Ispra, Italy
26-27 March, 2001

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ORGANISATION FOR ECONOMIC CO-OPERATION AND DEVELOPMENT

Pursuant to Article 1 of the Convention signed in Paris on 14th December 1960, and which came into force on 30th September 1961, the Organisation for Economic Co-operation and Development (OECD) shall promote policies designed:

- to achieve the highest sustainable economic growth and employment and a rising standard of living in Member countries, while maintaining financial stability, and thus to contribute to the development of the world economy;
- to contribute to sound economic expansion in Member as well as non-member countries in the process of economic development; and
- to contribute to the expansion of world trade on a multilateral, non-discriminatory basis in accordance with international obligations.

The original Member countries of the OECD are Austria, Belgium, Canada, Denmark, France, Germany, Greece, Iceland, Ireland, Italy, Luxembourg, the Netherlands, Norway, Portugal, Spain, Sweden, Switzerland, Turkey, the United Kingdom and the United States. The following countries became Members subsequently through accession at the dates indicated hereafter: Japan (28th April 1964), Finland (28th January 1969), Australia (7th June 1971), New Zealand (29th May 1973), Mexico (18th May 1994), the Czech Republic (21st December 1995), Hungary (7th May 1996), Poland (22nd November 1996), Korea (12th December 1996) and the Slovak Republic (14th December 2000). The Commission of the European Communities takes part in the work of the OECD (Article 13 of the OECD Convention).

NUCLEAR ENERGY AGENCY

The OECD Nuclear Energy Agency (NEA) was established on 1st February 1958 under the name of the OEEC European Nuclear Energy Agency. It received its present designation on 20th April 1972, when Japan became its first non-European full Member. NEA membership today consists of 27 OECD Member countries: Australia, Austria, Belgium, Canada, Czech Republic, Denmark, Finland, France, Germany, Greece, Hungary, Iceland, Ireland, Italy, Japan, Luxembourg, Mexico, the Netherlands, Norway, Portugal, Republic of Korea, Spain, Sweden, Switzerland, Turkey, the United Kingdom and the United States. The Commission of the European Communities also takes part in the work of the Agency.

The mission of the NEA is:

- to assist its Member countries in maintaining and further developing, through international co-operation, the scientific, technological and legal bases required for a safe, environmentally friendly and economical use of nuclear energy for peaceful purposes, as well as
- to provide authoritative assessments and to forge common understandings on key issues, as input to government decisions on nuclear energy policy and to broader OECD policy analyses in areas such as energy and sustainable development.

Specific areas of competence of the NEA include safety and regulation of nuclear activities, radioactive waste management, radiological protection, nuclear science, economic and technical analyses of the nuclear fuel cycle, nuclear law and liability, and public information. The NEA Data Bank provides nuclear data and computer program services for participating countries.

In these and related tasks, the NEA works in close collaboration with the International Atomic Energy Agency in Vienna, with which it has a Co-operation Agreement, as well as with other international organisations in the nuclear field.

COMMITTEE ON THE SAFETY OF NUCLEAR INSTALLATIONS

The NEA Committee on the Safety of Nuclear Installations (CSNI) is an international committee made up of scientists and engineers. It was set up in 1973 to develop and co-ordinate the activities of the Nuclear Energy Agency concerning the technical aspects of the design, construction and operation of nuclear installations insofar as they affect the safety of such installations. The Committee's purpose is to foster international co-operation in nuclear safety amongst the OECD Member countries.

CSNI constitutes a forum for the exchange of technical information and for collaboration between organisations which can contribute, from their respective backgrounds in research, development, engineering or regulation, to these activities and to the definition of its programme of work. It also reviews the state of knowledge on selected topics of nuclear safety technology and safety assessment, including operating experience. It initiates and conducts programmes identified by these reviews and assessments in order to overcome discrepancies, develop improvements and reach international consensus in different projects and International Standard Problems, and assists in the feedback of the results to participating organisations. Full use is also made of traditional methods of co-operation, such as information exchanges, establishment of working groups and organisation of conferences and specialist meeting.

The greater part of CSNI's current programme of work is concerned with safety technology of water reactors. The principal areas covered are operating experience and the human factor, reactor coolant system behaviour, various aspects of reactor component integrity, the phenomenology of radioactive releases in reactor accidents and their confinement, containment performance, risk assessment and severe accidents. The Committee also studies the safety of the fuel cycle, conducts periodic surveys of reactor safety research programmes and operates an international mechanism for exchanging reports on nuclear power plant incidents.

In implementing its programme, CSNI establishes co-operative mechanisms with NEA's Committee on Nuclear Regulatory Activities (CNRA), responsible for the activities of the Agency concerning the regulation, licensing and inspection of nuclear installations with regard to safety. It also co-operates with NEA's Committee on Radiation Protection and Public Health and NEA's Radioactive Waste Management Committee on matters of common interest.

FOREWORD

The Committee on the Safety of Nuclear Installations (CSNI) of the OECD-NEA co-ordinates the NEA activities concerning the technical aspects of design, construction and operation of nuclear installations insofar as they affect the safety of such installations.
The Integrity and Ageing Working Group (IAGE WG) of the CSNI deals with the integrity of structures and components, and has three sub-groups, dealing with the integrity of metal components and structures, ageing of concrete structures, and the seismic behaviour of structures. This workshop was proposed by the sub-group dealing with the seismic behaviour of structures.

Seismic re-evaluation is identified as the process of carrying out a re-assessment of the safety of existing nuclear facilities for a specified seismic hazard. This may be necessary when no seismic hazard was considered in the original design of the plant, the relevant codes and regulations have been revised, the seismic hazard for the site has been re-assessed or there is a need to assess the capacity of the plant for severe accident conditions and behaviour beyond the design basis. Re-evaluation may also be necessary to resolve an issue, or to assess the impact of new findings or knowledge.

In 1997, CSNI recognised the increasing importance of seismic re-evaluation for nuclear facilities throughout the world. It prepared a status report on seismic Re-evaluation NEA/CSNI/R(98)5 which summarized the current situation for Member countries of the OECD. The report suggested a number of areas of the seismic reevaluation process, which could be considered in the future. In May 2000, the seismic sub-group reviewed these suggestions and determined that it was timely to address progress on this topic through this workshop. The workshop focused on methods and acceptance criteria and, on countermeasures and strengthening of plant.

The workshop had 2 technical sessions listed below devoted to presentations, and a 3rd session devoted to a discussion of the material presented and to the formulation of workshop conclusions to update conclusions of the 1998 report.

<u>Session 1</u>

- Methods and acceptance criteria
- Benefits and disadvantages of the various methods of re-evaluation (Seismic PSA, Margins, deterministic, databases, tests …) in particular circumstances
- Role and scope of the peer review process
- Definition of the scope of the plant to be selected for the re-evaluation process
- Differences between re-evaluation and design criteria

<u>Session 2</u>

- Countermeasures/strengthening
- Civil engineering structures
- Post earthquake procedures and measures
- Strategies and priorities
- Recent innovation or research outputs

In the area of the seismic behaviour of structures, the CSNI is currently preparing among others a workshop on relations between seismological data and seismic engineering analysis to evaluate uncertainties and margins through a better description of real ground motion spectrum as opposed to a ground response design. Short reports on "lessons learned from high magnitudes earthquakes with respect to nuclear codes and standards" are under preparation and will cover several recent earthquakes.

Seismic reports issued by the group since 1996 are:

- NEA/CSNI/R(1996)10 Seismic shear wall ISP: NUPEC's seismic ultimate dynamic response test: comparison report, 1996. also referenced as: OCDE/GD(96)188

- NEA/CSNI/R(1996)11 Report of the task group on the seismic behaviour of structures: status report, 1997. also referenced as: OCDE/GD(96)189

- NEA/CSNI/R(1998)5 Status report on seismic re-evaluation, 1998.

- NEA/CSNI/R(1999)28 Proceedings of the OECD/NEA Workshop on Seismic Risk, CSNI PWG3 and PWG5, Tokyo, Japan 10-12 August 1999.

- NEA/CSNI/R(2000)2/VOL1 Proceedings of the OECD/NEA Workshop on the "Engineering Characterisation of Seismic Input, BNL, USA 15-17 November 1999 -

- NEA/CSNI/R(2000)2/VOL2 Proceedings of the OECD/NEA Workshop on the "Engineering Characterisation of Seismic Input, BNL, USA 15-17 November 1999

The complete list of CSNI reports, and the text of reports from 1993 onwards, is available on http://www.nea.fr/html/nsd/docs/

Acknowledgement

Gratitude is expressed to the European Commission Joint Research Centre, Ispra (VA), Italy for hosting the workshop as well as to the Organization for Economic Co-operation and Development (OECD) / Nuclear Energy Agency (NEA) / Committee on the Safety of Nuclear Installations (CSNI) / Integrity and Aging Working Group (IAGE) (Integrity of Components and Structures) for sponsoring our work.
Thanks are also expressed to chairmen of the sessions for their effort and co-operation.

The organizing Committee members were:

Mr John Donald, HSE (UK)
Mr Jean-Dominique Renard, Tractebel (B)
Dr Vito Renda, JRC/ISPRA (I)
Prof. Pierre Labbé, IAEA
Dr. Tamas Katona, PAKS (HU)
Dr Andrew Murphy, USNRC (USA)
Mr Eric Mathet, OECD/NEA

OECD/NEA WORKSHOP ON THE SEISMIC RE-EVALUATION OF ALL NUCLEAR FACILITIES

26-27 March 2001
Ispra, Italy

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B. PROGRAMME

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SESSION 1:		**Methods and Acceptance criteria**	
		Chairman:	**Mr. J.D Renard - Tractebel(B)**
9:40	**10:00**	The IAEA Safety Report of seismic re-evaluation of existing NPPs	*Pierre Labbé , IAEA*
10:00	**10:20**	Individual Plant Examination of External Events (IPEEE):Seismic Analyses Methods and Insights	*Charles Hofmayer, BNL (USA)*
10:20	**10:40**	Seismic Re-evaluation of Kolzoduy NPP: Criteria, Methodology, Implementation	*Dr. Marin Kostov - Risk Eng Ltd, (Bulgaria)*
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11:40	**12:00**	Seismic Margin Assessment of Spanish Nuclear Power Plants: a Perspective from Industry and Regulators	*Francisco Beltrán - IDOM (SP)*
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15:20	**15:40**	Damage Detection and Assessment by Ambient Vibration Monitoring	*Dr. Helmut WenzeL - VCE (A)*
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17:40	**18:00**	Re-Assesment of Seismic Safety of TR-2 Research Reactor Building	(Paper included but not presented) *Ayhan Altinyollar - TAEA (TR)*
18:00	**18:20**	Seismic Proving Test of Concrete Containment Vessel	*Hattori Kiyoshi - NUPEC (J)*
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Tuesday March 27, 2001

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9:00	**9:15**	Seismic Hazard Re-evaluation in Armenian NPP Site	*S.Balassanian - NSSP (Armenia)*
9:15	**9:30**	Seismic Hazard of the Korean NPP Sites: Recent Innovation in the R&D and Hazard Result	*JEONG-MOON, SEO- KAERI (KR)*
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9:45	**10:00**	French operating NPP Design floor response spectra: margins related to soil-structure interaction	*Stéphane Vallat - EDF (F)*
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		Chairman:	**Mr. J. Donald - HSE (UK)**
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11:10	**11:30**	Seismic safety re-evaluation and enhancement at the Paks NPP	*Dr. Tamás Katona - Paks NPP (HU)*
11:30	**11:50**	SEISMIC RE-EVALUATION PROGRAM OF THE ARMENIA NUCLEAR POWER PLANT – Results from an international co-operation project	*Dr Lamberto d'Andrea - SOGIN (I)*

11:50	**12:10**	The Seismic Assessment of British Energy's Nuclear Power Stations and some Pragmatic Solutions to Seismic Modifications.	*John P McFarlane - British Energy (UK)*
12:10	**12:30**	Intercomparison of analysis methods for seismically isolated nuclear structures	*K.N.G. Fuller - TARRC (UK)*
<u>**12:30**</u>		<u>Lunch</u>	

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		Chairman:	**Mr. P. Sollogoub - CEA (F)**
15:00	**16:30**	**Panel discussion**	
16:30		Closure	

C. PAPERS

SESSION 1
METHODS AND ACCEPTANCE CRITERIA
Chairman: J. D. Renard

Seismic Re-evaluation of Existing Nuclear Power Plants

An Introduction to an IAEA Safety Report

Pierre Labbé

International Atomic Energy Agency, Vienna.

ABSTRACT

The purpose of this paper is to introduce the IAEA Safety Report under preparation on the Seismic Reevaluation of existing Nuclear Power Plants (NPPs). After an introduction on the objectives of the IAEA document and the common technical background of seismic re-evaluation, the outlines of the seismic reevaluation process are discussed: the review level earthquake, the safety analysis of the plant as it should be considered in view of a seismic re-evaluation, the use of the feedback experience and the practice of walkdown. A special emphasize is given to the capacity evaluation, including some recommendations about non linear analyses and consequences of non linear behaviour on the seismic response. It is proposed to account for post elastic behaviour by the way of "Energy Absorption Factors", provided actual ductile capacity is available.

INTRODUCTION

Objective of the IAEA document

The main purpose of the IAEA Safety Report under preparation is to provide guidance for conducting a seismic safety evaluation programme for an existing nuclear power plant in a manner consistent with current criteria and internationally recognized practice.

The document may be a tool for regulatory authorities and responsible organizations for the execution of the seismic safety evaluation programme, giving a clear definition to different parties, organizations and specialists involved in its implementation on
(i) objectives of the seismic evaluation programme:
(ii) phases. tasks and priorities in accordance with specific plant conditions;
(iii) a common and integrated technical framework for acceptance criteria, and capacity evaluation.

Technical findings

Evaluate the safety of an existing NPP against earthquakes is a more complicated task than safely design a new NPP for the same purpose, and an appropriate evaluation program is not easy to set up. The IAEA Safety Report recalls main technical findings that should be regarded as a common background and should therefore guide the set-up of a seismic evaluation program. Among them, we notice the following ones.

Despite peak ground acceleration (PGA) is a parameter widely used in order to scale the seismic input. it is also a known technical finding that damaging capacity of seismic input motion is poorly correlated to the PGA level, even the elastic response spectrum is a poor tool for that. Other parameters such as duration play a significant role in a judicious evaluation. Consistently, it is known that near field earthquakes with small or moderate magnitudes can lead at once to significant PGA levels and to non significant damages to structures; on the other hand they may lead to spurious behaviour of instrumentation and control systems.

Regarding the structures and the mechanical components, it is a result of research and development in the past decade that, due to the usually available ductility and to the dynamic aspects of the phenomenon, a safe anti-seismic design relies more on capacities in accommodating large strains than in balancing large forces as they can be estimated on the basis of the classical engineering approach (elastic behaviour assumption and a static equivalent approach). It is the reason why a special attention has to be paid to actual ductile capacities.

OUTLINES OF A SEISMIC RE-EVALUATION

Review level earthquake

The assessment of the seismic hazard of the site should evaluate the ecological stability- of the site. e.g. the absence of any capable fault that can produce differential ;round displacement phenomena underneath or in the close vicinity of buildings and structures important to seismic safety. On the principle the re-evaluation process should not reveal dramatic changes in this field. Regarding the severity of the seismic ground motion, it may likely differ from the one at the design stage. It has to be described with appropriate parameters such as peak ground acceleration. velocity, and/or displacement ground response spectra; duration. time history accelerograms, etc.

The IAEA Safety Report points out that the results of non linear time history analyses are very sensitive to the choice of the input motion. When such analyses cannot be avoided (in some geotechnical issues for instance), the IAEA Safety Report insists on the choice of the accelerograms, which has to be carried out very carefully.

Either Safety Margin Assessment (SMA) or Probabilistic Safety Assessment (PSA) are proposed by the IAEA. In the SMA framework. a Review Level earthquake (RLE) is introduced. On the principle, it should be at least at the level of. the S1.2 earthquake (as defined in the corresponding, lAEA Safety Guide[1]). The S1.2 level itself should be updated in case any reason for that appeared since the evaluation of the SL.-2 design level.

Safety analysis

The purpose of the Safety Analysis is to determine the Selected Structures Systems and Components (SSSC) required for a seismic evaluation ,and to specify for each of them what are the required functions they have to assure, or what are the failure modes that have to be prevented from, during or after an earthquake.

It is not necessary to verify the seismic adequacy of all plant equipment defined as Seismic Category I for the design of new facilities in IAEA Safety Guide 50-SG-D15 [2]. In the SMA method it is common international practice to focus the evaluation only oft a list of Selected Structures, Systems and Components (SSSC (e.g. Structures. mechanical and electrical items, 1 & C. distribution systems, components) essential to bring the plant from a normal operation condition to safe shutdown conditions and to ensure safety during and following_the occurrence of a RLE. The objectives are to identify seismic vulnerabilities. if any. which. if remedied, will result in the plant being able to shut down safely in the event of such earthquake.

Because there is redundancy and diversity in the design of nuclear power plants. there may be several paths or trains which could be used to accomplish the required safety functions. as a minimum condition. only the active and passive equipment in a primary path (or train) and back up equipment within that path and a backup path need to be identified for seismic evaluation purpose to the RLE. The preferred safe shutdown path should be selected and clearly indicated. In selecting the primary and attenuate shutdown paths, a single active failure must not lead to a significant increase in damage core frequency.

Most of the criteria and assumptions developed for the margin method are equally applicable to the PSA method. The primary difference is that in the PSA method the list of SSSC to be reviewed is based on the results of the PSA plant systems analysis. The internal event and fault trees are modified to include spatial interactions. failure of passive components such as structures and supports, and common-cause effects of seismic excitation. A detailed discussion of the interpretation of seismic PSA results can be found in IAEA Tec. Doc. 724 [3]

The SSSC list may be expanded to include additional components as requested or required by the owner, operator, licensee or the regulator. A typical example of expanded scope is cooling of the spent fuel pool.

Failure modes

For each SSSC. the required functions it has to assure has to be specified. Fur instance: a) For a structure it should be specify whether stability or Functionality (supporting of equipment) is required. Due consideration should be given to structural elements required for fulfilling leak tightness requirements. b) For mechanical components , those which should keep their integrity. and those which should remain operable should be listed.

At this stage. it is necessary to develop a .clear definition of what constitutes failure for each of SSSC being evaluated. Several modes of seismic failure may have to be considered. Identification of credible failure modes is based largely on the feedback experience and judgement of the reviewers. In this task. a review of the performance of similar structures, systems and components and of reported failures in industrial facilities subjected to strong motion earthquakes will provide useful information. In these regards, the IAEA Safety Report reminds the most frequently observed failure modes. Likewise, consideration of design criteria, qualification test results. calculated stress levels in relation to allowable limits and seismic fragility evaluation studies done on other plants will prove helpful.

Feedback experience

The IAEA Safety Report recognizes that seismic evaluation of existing NPPs rely much more on feedback experience than assessment of new NPPs does. Estimate of seismic capacity of systems and components may often be accomplished by the use of experience gained from real strong motion seismic events. Such qualification requires that:

a) the seismic excitation of an item installed in a plant subjected to a real strong motion earthquake effectively envelops the seismic input motion defined for similar items at the given NPP:
b) the item being evaluated and the one which underwent the real strong motion earthquake have similar physical characteristics and have similar support or anchorage characteristics (alternatively. the support or anchorage capacities can be evaluated by additional analysis):
c) in the case of active items, the item subjected to the strong motion earthquake performed similar functions during or following that earthquake. including the potential aftershock effects, as would be required for the safety related item being evaluated.

Use of feedback experience has to be made in the framework of a validated procedure associated to an appropriate database. The IAEA Safety Report regards the Generic Implementation Procedure (GIP) developed by the Seismic Qualifications Utility Group (SQUG) jointly with the NRC as an example of such a procedure [4].

However. it is pointed out that most building structures and some systems and components are so specialized that they are not included in the earthquake experience database. Particularly, it is the. case of components and structures and major pieces of equipment of the reactor coolant.

Walkdown

The procedure associated to the use of feedback experience implies the practice of walkdown. Major objectives of a walkdown are:

a) to review the SSSC: to confirm the list of the SSSC, their required functions. their possible failure modes: to screen out the SSSC which feature a seismically robust construction and to identified the easy-fixes that have to be carried out regardless any analysis; to confirm that the database is appropriate to the SSSC under consideration,
b) to check the extent to which the as-built conditions correspond to design drawings when the evaluation is based on analysis.
c) to define representative sample configurations for limited analytical evaluations.
As the basis for a plant seismic walkdown, the SSSC list should be prepared in advance. indicating the required functions to be assured. During a walkdown. a special.care should be paid to spatial interaction and to anchorage of equipment. Each SSSC should be visually examined. After a preliminary screening walkdown, there will be three alternative disposition categories for each SSSC:

a) the seismic capacity is not adequate. so a modification is required:
b) the seismic capacity is uncertain and further evaluation is needed to determine whether a modification is required. or
c) the seismic capacity is adequate for the specified RLE .

General experience with plant walkdown have indicated that most electrical and instrument cabinets require modifications to increase the anchorage capacity and many unreinforced masonry walls also require upgrades. Electrical and mechanical distribution system supports have required selective upgrading. Some mechanical equipment require upgrading of their anchorage capacity.

EVALUATION OF SEISMIC: MARGIN CAPACITY

Consequences of post-elastic behaviour

The evaluation process basically leads to deal with post elastic behaviour. Nevertheless to the possible extent, it is recommended by the IAEA Safety Report to avoid sophisticated controversial non-linear analyses. In order to make a judgement. it is highly preferable to document about the "as is" facilities relevant data that support a simple analysis than to provide a large amount of non linear analyses. However, it is recognize that static non-linear analyses (such as the "push-over" method) may be of interest to assess the margins of a structure or of a mechanical system.

According to the state of the art, due to the post elastic behaviour, the purpose of the evaluation of seismic margin capacities should be to analyse the strains induced by the postulated RLE in the Structure and to compare them to ultimate admissible strains. Basically, it means that approaches orientated to strain evaluation (displacements approach) are more relevant that those based on stresses evaluation (forces approach).

Consistent strain analysis is generally difficult to obtain because engineering practices and engineering_tools (education. standards, criteria. computer codes...) are orientated towards stresses analysis. For this reason, in order to provide convenient guidance, the IAEA Safety Report is written in the general framework of stresses analysis; in this framework the inelastic energy absorption factor Fy is introduced.

Response analysis

In computing the response of the structure. the following principles are recommended:

a) a reference model of the structure. including soil-structure interaction effects, should be derived from a best estimate approach without intentional conservative bias (for instance values of structural damping higher than for design are accepted), however

b) parametric studies have to be carried out in order to cover the uncertainties of the model. Particularly the variability of the soil profile Young modulus is recommended to be at least in a range from 0.67 to 1.5 times the best estimate, preferably from 0.5 to 2 times. Also the fragility of masonry wall is pointed out. which should lead to parametric studies.

In any case the range of the natural frequency to be taken into account for the parametric study, and accordingly for the description of the floor response spectra should be so that

$$\bullet f = f_{max} = f_{min} < 0.2 f_{best\ estimate}$$

This range of variation is centered on the best estimate in case of elastic analysis and shifted to the low frequency in case F• factors are used in the capacity analysis. In case F• factors above a given threshold are used the dynamic model of the structure has to be updated and the response accordingly computed.

Capacity evaluation

The IAEA Safety Report recommends that the general criteria for the assessment of the seismic margin capacity be more conservative than those which would be permitted in conventional seismic design but more liberal than in original nuclear power plant design. The above mentioned F• factors are calibrated accordingly.

The F• factors are introduced as a practical inclusive method that takes into account the ductile capacity of the structures. in this context. a is the admissible ductile capacity. In this regards, the LAEA Safety Report insists on the fact that an actual ductile capacity has to be available and requires that any use of F• factors be documented, even roughlv. The objective is to be sure that any brittle failure mode is impossible or has been duly addressed, and that the engineers in charge of the re-evaluation have a good knowledge of the structure and components they are in charge of assessing.

Values of F• larger than those proposed in the Safety Report are permitted, provided they are supported by an appropriate documentation including experimental evidences and analytical background (such as the displacement orientated approach), and a consistent analysis process is used for estimating the response of the structure or components considered.

In order to illustrate the order of magnitude of the F• factors, table I gives some typical values that were used in seismic re-evaluation of existing NPPs. It has to be pointed out that the use of such value: is consistent only with the use of the associated values of parameters that govern the seismic response, such as damping.

Table 1. Some Typical F• values used in seismic re-evaluation of existing NPPs

Concrete columns where flexure dominates	l.25 to 1.50
Concrete columns where shear dominates	1.00 to 1.25
Concrete connections	1.00 to l.25
Concrete shear walls	1.50 to 1.75
Steel columns where shear dominates	1 00 to 1.25
Steel beams where flexure dominates	1.50 to 1.75
Welded steel pipes	1.50 to 2.00

The principle for the use of the F• factors is the following one: first an elastic behaviour analysis of the structure is carried out. Then the elastic stresses resulting from this analysis are divided by the F• floor before entering the classical criteria for stress analysis. In case of stresses resulting from a displacement controlled load (seismic differential anchor motion), the elastic stresses are divided by •. Basically this rule means that dividing either by F• or by •, the intent is to account for the primary part of the considered stresses.

On the other hand elastically computed displacements arc amplified by the F• factor. According to a common view, elastically computed displacements are slightly modified in case of post elastic behaviour. In this regard the IAEA rule is intentionally more severe than the common practice, taking into account a lesson of the feedback experience: many observed damages are due to an underestimate in the design phase of the displacements induced by an earthquake.

Formula governing the use of the F• factors are given in the appendix.

Special items

The IAEA Safety Report states that special care has to be taken with two items:

a) Anchorage capacities, since anchorages are well known as a weak point of industrial facilities in case of earthquake: specific criteria are mentioned.

b) Functional capacities of electromechanical relays. since feedback experience draws attention to their frequent spurious alarms.

EXAMPLES OF SEISMIC RE-EVALUATION OF NPPs

Several eastern Europe NPPs proceeded to seismic re-evaluation in the last years. These re-evaluations were carried out on the: basis of Guidelines that were reviewed by the IAEA. So it is possible to say that in spite the Safety Report itself is just to be issued. its principles have been already extensively used in seismic re-evaluations of existing NPPs. Furthermore it is also used for re-evaluation of sites with other facilities than NPPs. The table 2 indicates some values of Review Level Earthquakes.

Table 2. Examples of Review Level Earthquake

Armenian NPP	0.35g
Bohunice. Slovakia	0.35g
Paks. Hungary	0.25g
Kozloduy, Bulgaria	0.20g

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APPENDIX : USE OF THE F• FACTORS

For a primary structure:

S'_p: inertial stresses computed assuming an elastic behaviour of the structure, with around motion RILE as input.

S'_p S_p reduced by the appropriate $F•_p$ value of this primary structure, as follows
$S'_p = S_p/F•_p$

D'_p displacements computed assuming an elastic behaviour of the structure. with ground motion RLE as input.

D'_p D_p amplified by the appropriate $F•_p$ value of this primary structure, as follows
$D'_p - D_p F•_p$

For a secondary structure (piping systems , components, ducts...), two types of input motion have to be provided: floor response spectra and anchor displacements; they have to be consistent. i.e. they have to be computed with the same model of the primary structure. In most of the cases. anchor input motion have to be taken into account to the extent they result in, non nil differential displacements.

S_s: inertial stresses computed assuming an elastic behaviour of the secondary structure. with FR'_p, as input.

S'_s S'_s reduced by the appropriate $F•_p$ value of this secondary structure, as follows
$S'_s = S_{s/}F•_s$

S_m stresses induced by anchor motion eoroputed assuming an elastic behaviour of the secondary structure, with D'_p motion of the primary structure as anchor input motion.

S'm S_m modified as follows (primary part of a displacement controlled load):
$S'_m = S_m/•_s$

In case the displacements in this secondary structure have to be estimated (for instance displacements of a run pipe are anchor input motion for a connected branch pipe), the following formula can be used:

D_s displacements computed assuming an elastic behaviour of the secondary structure, with FR'_p and D'_p as input motions.

D'_s D_s amplified by the appropriate F• value of this secondary structure, as follows:

$$D'_s = D_s F•_s$$

The S' stresses are used in superposition with stresses induced by permanent loads in order to verity compliance with seismic capacity: criteria.

The U.S. Nuclear Regulatory Commission Perspective on Seismic Re-evaluation of Nuclear Power Plants: The IPEEE Methodology

Roger M. Kenneally, Nilesh C. Chokshi, and Andrew J. Murphy
U.S. Nuclear Regulatory Commission
Office of Nuclear Regulatory Research
Washington, DC 20555-0001, USA

ABSTRACT

Since the late 1970's, seismic re-evaluation has been carried out in several stages at the U.S. Nuclear Regulatory Commission (USNRC). In the late 1970's and early 1980's, the Systematic Evaluation Program (SEP) was carried out to re-evaluate the eleven oldest nuclear power plants which received their construction permits between 1956 and 1967. In 1980, I&E Bulletin No. 80-11, was published requesting that licensees evaluate masonry walls whose failure could affect safety-related systems. In the late 1980's, Unresolved Safety Issue (USI) A-46 was initiated to verify the seismic adequacy of mechanical and electrical equipment in many of the older plants. Affected plants were typically those whose construction permit application was docketed (submitted) before about 1972. In the early 1990's the USNRC initiated the Individual Plant Examination of External Events (IPEEE) Program; seismic was one of the major external events that had to be included in the licensee's evaluation.

This paper focuses on the seismic portion of the IPEEE analyses. It discusses acceptable methods for performing the seismic evaluation, enhancements to these methods that reflected the state-of-the-art improvements (circa late 1980's - early 1990's). [Note: Guidance reflecting the current (circa 2000) state-of-knowledge for performing a seismic PRA or seismic margins assessment can be found in ANS, 2000. The NRC has not developed a position on this document.]

INTRODUCTION

Based on USNRC and industry experience with plant-specific probabilistic risk assessments (PRAs), the USNRC recognized that systematic examinations are beneficial in identifying plant-specific vulnerabilities to severe accidents, generally, beyond design basis considerations. As part of the implementation of the policy statement on severe accidents in nuclear power plants [USNRC, 1985], the USNRC issued Generic Letter 88-20 in November 1988 [USNRC, 1988], requesting that each licensee conduct an Individual Plant Examination (IPE) for internally initiated events only. The USNRC staff needed time to identify which external hazards needed to be evaluated, and to identify acceptable examination methods and develop procedural guidance. Therefore, the USNRC did not issue until June 1991, Generic Letter 88-20, Supplement 4 [USNRC, 1991a], requesting a systematic individual plant examination for severe accidents initiated by external events. Procedural and submittal guidance for conducting the IPEEEs is provided in NUREG-1407 [USNRC, 1991b]. Basically, two methodologies are considered acceptable to identify potential seismic vulnerabilities at nuclear power plants. The first is a seismic probabilistic risk assessment (PRA), the second is the seismic margins methodology. Table 1 provides a comparison of seismic PRA and the two seismic margins methodologies. Table 2 describes the advantages and disadvantages with the seismic margins methodology. Table 3 describes the advantages and disadvantages with the seismic PRA methodology [Tables 1-3 based on Shao et al, 1990,and ASCE, 1998]. Subsequent to the publication of NUREG-1407 [USNRC 1991b], the USNRC issued Supplement 5 to Generic Letter 88-20 [USNRC, 1995], to notify licensees of modifications to the recommended scope of the seismic portion of the IPEEE for certain sites in the central and eastern U.S.

IPEEE OBJECTIVES

The objectives of the IPEEE, which are similar to the objectives of the internal event IPE, are for each licensee (1) to develop an appreciation of severe accident behavior, (2) to understand the most likely severe accident sequences that could occur at a licensee's plant under full power conditions, (3) to gain a qualitative understanding of the overall likelihood of core damage and fission product releases, and (4) if necessary, to reduce the overall likelihood of core damage and radioactive material releases by modifying, where appropriate, hardware and procedures that would help prevent or mitigate severe accidents. In meeting the objectives of the IPEEE, the examination should focus on qualitative insights from a systematic plant examination rather than only on absolute core damage frequency estimates.

ACCEPTABLE METHODOLOGIES AND ENHANCEMENTS FOR IPEEE ANALYSES

The following sections summarize the guidance and enhancements developed by the USNRC for licensees performing a new seismic PRA, updating an existing seismic PRA, or using the seismic margins methodology to identify potential seismic vulnerabilities at nuclear power plants.

Seismic PRA

The PRA should be at least a Level 1 plus containment performance analysis. The basic elements of a seismic PRA are: (1) hazard analysis, (2) plant system and structure response analysis, (3) evaluation of component fragilities and failure modes, (4) plant system and sequence analysis, and (5) containment and containment system analysis including source terms, to identify unique seismic sequences or vulnerabilities different from the internal event analysis.

New Seismic PRA Analysis

The following summarizes the guidance and enhancements in USNRC, 1991b, provided to licensees that were planning to use a seismic PRA for their IPEEE analyses:

General Considerations. Organizations planning to do both an internal events PRA and a seismic PRA should be aware of important considerations that, if incorporated in the planning of the internal events PRA , will minimize their resource expenditures. For example, (1) a well-organized walkdown team and a properly planned walkdown will enable many issues to be addressed at the same time; (2) the peer review team should consider the need to review both internal and external event analyses; (3) fault tree analysts for internal events should be aware of spatial interactions (including internal flooding effects), failure of passive components such as structures and supports, and common-cause effects (the culling or pruning of trees should be done with these considerations in mind); and (4) internal event models should be developed knowing that, in the seismic analysis, the fragilities of a component are sensitive to elevation. Also, a component and its peripheral equipment may have different fragilities.

PRA calculations that account for all uncertainties are clearly acceptable. However, for purposes of the IPEEE, it is not necessary to carry out complete uncertainty quantifications defining a distribution of core-damage frequencies in order to identify vulnerabilities. Mean point estimates using a single hazard curve (rather than a family of hazard curves) and a single fragility curve (rather than a family of fragility curves) for each component are sufficient to get insights into potential seismic vulnerabilities. This point estimate approach is valid only because of the limited IPEEE objective: to identify dominate sequences and components and where possible rank them.

Hazard Selection. For the central and eastern U.S., two highly sophisticated seismic hazard studies were conducted by the Lawrence Livermore National Laboratory (LLNL) and the Electric Power Research Institute (EPRI). For many sites, these studies yielded significant differences at the low probability and high-level ground motions. While a full seismic hazard uncertainty analysis is not necessary in performing a seismic PRA for the IPEEE, mean (arithmetic) hazard estimates from both studies should be used to obtain two different point (mean) estimates. If only one analysis is used, it should be the higher of the two mean (arithmetic) hazard estimates. The use of both mean hazard curves has another advantage in that the extent of uncertainty will become obvious and the emphasis on the bottom line numbers is reduced.

Seismic PRA Methodology Enhancements.

Plant Walkdowns. Walkdowns are performed to find as-designed, as-built, and as-operated seismic weaknesses in plants. Walkdowns should be consistent with the Sections 5 and 8, and Appendices D and I of EPRI NP-6041, 1988.

Relay Chatter. Relays in this context, include components such as electric relays, contactors, and switches that are prone to chatter. The complexity of the evaluation should be consistent with the site's seismic margin review level earthquake. [The review level earthquake and its use are described in the section, "Seismic Margin Methodology."]

Liquefaction. The potential for soil liquefaction and associated effects on the plant need to be examined for some sites because of specific site conditions. Procedures are described in EPRI NP-6041.

Use of an Existing Seismic PRA Analysis

The following summarizes the guidance and enhancements in USNRC, 1991b, provided to licensees that were planning to update an existing seismic PRA for their IPEEE analyses:

General Considerations. The use of an existing seismic PRA to address the seismic IPEEE is acceptable provided the PRA reflects the current as-built and as-operated condition of the plant, and the deficiencies of past PRAs are adequately addressed.

Seismic PRA Methodology Enhancements.

Hazard Selection. For PRAs at central and eastern U.S. plant sites that did not use the LLNL or the EPRI mean hazard estimates, sensitivity studies should be conducted to determine if the use of these results would affect the delineation or ranking of seismic sequences. For PRAs in the western U.S., sensitivity studies should be carried out to determine the effect of uncertainty in the hazard on the delineation and ranking of seismic sequences.

Walkdowns. Since a walkdown is considered to be one of the most important ingredients of the seismic IPEEE, a supplemental walkdown in conformance with the intent of the procedures described in Sections 5 and 8, and Appendices D and I of EPRI NP-6041, should be performed. It may be necessary to amplify the earlier analysis based on the walkdown outcome.

Relay Chatter. See the discussion for a new PRA.

Liquefaction. See the discussion for a new PRA.

Nonseismic Failures and Human Actions. In several seismic PRAs, nonseismic failures and human actions have been important contributors to seismically induced core damage frequencies or risk indices. Examples of nonseismic failures include failures of the auxiliary feedwater system and failure of feed and bleed mode of cooling, battery depletion, and power operated relief valve failures. Examples of human actions include delays or failures in performing specified actions or operator misdiagnoses a situation and takes improper action that is not related to the actual, current plant situation. The analyst has the option to expand its PRA or demonstrate that the exclusion of nonseismic failures and human actions will not significantly alter the PRA results or insights.

Seismic Margin Methodology

The following summarizes the guidance and enhancements in USNRC, 1991b, provided to licensees that were planning to use the seismic margins methodology for their IPEEE analyses.

Two methodologies are currently available: one developed under USNRC sponsorship and the other developed under EPRI sponsorship. The two methods use different systems analysis philosophies. The USNRC method [NUREG/CR-4334, 1985; NUREG/CR-4482, 1986; NUREG/CR-5076, 1988] is based on an event/fault tree approach to delineate accident sequences. The EPRI methodology [EPRI NP-6041, 1988] is based on a systems "success path" approach. This approach defines and evaluates the capacity of those components required to bring the plant to a stable condition (either hot or cold shutdown) and maintain that condition for at least 72 hours. Several possible success paths may exist. Both methods require an evaluation of containment performance.

General Considerations. Each analyst should examine its plant critically to ensure that the generic insights used in the margin methodology development to identify critical functions, systems, and success path logic are applicable to its plant. This is particularly vital for older plants where systems and functions may differ greatly from the plants considered in the development of the margins methodologies.

Review Level Earthquake. The seismic margin methodology was designed to demonstrate sufficient margin over the Safe Shutdown Earthquake (SSE) to ensure plant safety and to find any "weak links" that might limit the plant shutdown capability to safely withstand a seismic event larger than the SSE or lead to seismically-induced core damage. The methodology involves the screening of components based on their importance to safety and seismic capacity. The methodology utilizes two review or screening levels geared to peak ground accelerations of 0.3g and 0.5g (0.8g and 1.2g spectral acceleration). In areas of low to moderate seismic hazard, most plants that have been evaluated using seismic PRAs or seismic margin studies have been shown to have high-confidence, low probability of failure (HCLPF) values at or below 0.3g. Past experience indicates that, at the 0.3g screening level, a small number of "weak links" are likely to be identified, efficiently defining the dominant contributors to seismically-induced core damage. It is the USNRC staff's judgement that the use of a 0.3g review level earthquake for most of the nuclear power plants sites in the central and eastern U.S. (East of the Rocky Mountains) would serve to meet the objectives of the IPEEE. However, all sites in the central and eastern U.S. are not subject to the same level of earthquake hazard. For some sites where the seismic hazard is low, an approach centered on walkdowns is acceptable. For western sites other than California coastal sites, a 0.5g review level earthquake should be used.

USNRC, 1991b, describes the procedure used by the staff to group the plant sites into one of the following bins: Reduced Scope, 0.3g Focused Scope, 0.3g Full Scope, 0.5g Full Scope, and Seismic PRA. In summary, hazard comparisons were made using the mean, median and 85th percentile from the site-specific results provided by the LLNL and EPRI studies. Each of these pieces of information represents a different way of characterizing the distribution of seismic hazard estimates at each site as determined by a particular study. In addition, site hazard comparisons were made using response spectra and peak ground acceleration. The likelihoods of exceeding spectral response accelerations in the 2.5 to 10 Hz range were examined because these frequencies are more closely related to the types of motion that could cause damage at nuclear power plant sites. Weights were assigned to the likelihoods of exceeding spectral response ordinates at 2.5, 5, and 10 Hz, and the peak ground acceleration. Emphasis was placed on the relative ranking of sites with respect to other sites using the same seismic hazard study, statistic, and ground motion measures.

The USNRC staff has designated nuclear power plant sites into the following seismic evaluation categories:

Eastern U.S. Plant Sites	**Western U.S. Plant Sites**
Reduced Scope (9 sites)	
0.3g Focused Scope (50 sites)	
0.3g Full Scope (5 sites)	0.3g and 0.5g Full Scope (2 sites)
Seismic PRA (2 sites committed)	Seismic PRA (2 sites)

<u>Scope of the Evaluation</u>

Reduced Scope. For sites where the seismic hazard is low, a reduced-scope seismic margins method emphasizing the walkdown is adequate. Well-conducted, detailed walkdowns have been demonstrated to be the most important tool for identifying seismic weak links whose correction is highly cost effective.

Focused Scope, Full Scope (0.3g or 0.5g). Plants have similar evaluation scopes, but the Focused Scope submittals involve a more limited evaluation of soil failures, relay chatter, and component capacities.

Subsequent to the publication of USNRC 1991a and 1991b, the USNRC issued Supplement 5 to Generic Letter 88-20 [USNRC, 1995], to notify licensees of modifications to the recommended scope of the seismic portion of the IPEEE for certain sites in the central and eastern U.S.

Containment Performance

The primary purpose of the containment performance evaluation is to identify sequences and vulnerabilities that involve containment, containment functions, and containment systems seismic failure modes or timing that are significantly different from those found in the IPE internal events evaluation.

Containment Penetrations. Generally, containment penetrations are seismically rugged; a rigorous fragility analysis is needed only at seismic margin review levels greater than 0.3g, but a walkdown to evaluate for unusual conditions (e.g., spatial interactions, unique configurations) is recommended. An evaluation of the backup air system of the equipment hatch and personnel lock that employ inflatable seals should be performed at all review level earthquakes. Also, some penetrations need cooling, and the possibility and consequences of a cooling loss caused by an earthquake should be considered.

Valves. Valves involved in the containment isolation system are expected to be seismically rugged. A walkdown to ensure that they are similar to test data and have known high capacities, and that there are no spatial interaction issue will suffice. Seismic failures of actuation and control systems are more likely to cause isolation system failures and should be included in the examination. For valves relying on a backup air system, the air system should also be included in the seismic examination.

Heat Removal/Pressure Suppression Functional System. Components of the containment heat removal/pressure suppression functional system that are not included elsewhere and are not known to have high capacities should be examined. An example of such a component might be a fan cooler unit supported on isolator shims. The walkdown should include examination of such components and their anchorage. Similarly, support systems and other system interaction effects (e.g., relay chatter) should be examined.

STATUS

The NRC staff and its contractors have nearly completed their reviews of 70 seismic IPEEE submittals. The preliminary observations are available in a paper presented at the PSAM 5 Conference [Rubin et al, 2000]. Similar to what was done at the conclusion of the IPE review, the NRC staff has published a draft report for comment entitled, “Perspectives Gained from the Individual Plant Examination of External Events (IPEEE) Program,” [USNRC, 2001].

One of the significant observations from the reviews performed to date is “Regardless of the specific approach used, all performed a detailed seismic walkdown, and many of the insights gained by licensees resulted from the walkdowns.” [Rubin et al, 2000]. This observations comports well with the NRC staff expectations during the development of the seismic IPEEE guidance.

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DISCLAIMER

This paper was prepared in part by employees of the United States Nuclear Regulatory Commission. It presents information that does not currently represent an agreed-upon staff position. USNRC has neither approved nor disapproved its technical content.

Table 1. Comparison of Seismic PRA and Seismic Margins Methodologies

Seismic PRA	USNRC Seismic Margins Method (as modified by USNRC, 1991b)	EPRI Seismic Margins Method (as modified by USNRC, 1991b)
Approach		
Probabilistic	Semi-probabilistic	Partially probabilistic
Scope of Review		
Event trees and fault trees are usually developed from the event/fault trees developed for the internal events analyses. Structures and elements where failure could impact and fail safety-related elements are added to the trees	For pressurized water reactors, the safety functions of reactor criticality and early emergency core cooling are considered. For boiling water reactors, the safety functions of reactor subcriticality, emergency core cooling, and residual heat removal are considered. In addition, a small break loss-of-coolant accident (LOCA) is postulated to occur, and soil failure modes are considered. Potential for earthquake-induced flooding and earthquake-induced fires is also considered, as well as nonseismic failures and human actions.	Review includes electrical, mechanical, and NSSS equipment; piping; tanks; heat exchangers; cable trays and conduit raceways; containment; and structures. In addition, leakage equivalent to a small break LOCA is postulated to occur in one success path, and soil failure modes are considered. Potential for earthquake-induced flooding and earthquake-induced fires is also considered, as well as nonseismic failures and human actions.
Seismic Input		
Site-specific hazard curves, for instance, those developed by the Electric Power Research Institute or the Lawrence Livermore National Laboratory, for peak ground acceleration and response spectra should be used.	A response spectrum shape is specified, for instance, NUREG/CR-0098 median shape, anchored to 0.3g or 0.5g, as appropriate, may be used. Development of new instructure response spectra, including effects of soil-structure interaction, is encouraged.	Same as the USNRC seismic margins method.
Selection of Equipment		
Elements whose failure could lead to core damage (i.e., Level 1 PRA) are considered initially. Fault trees are "pruned" based on systems and fragility considerations.	Elements whose failure could lead to core damage are considered initially. Fault trees are "pruned" based on systems and fragility considerations.	Two separate and independent shutdown success paths are selected. One path postulates leakage equivalent to a small break LOCA.

Table 1. Comparison of Seismic PRA and Seismic Margins Methodologies (cont)

Screening Requirements		
Screening based on system and fragility considerations.	In general, equipment functionality is investigated based on seismic experience or test data. Equipment anchorage is analyzed for each component. Caveats and guidance are provided in the criteria screening tables in NUREG/CR-4334 and EPRI NP-6041 for three ranges of seismic input.	In general, equipment functionality is investigated based on seismic experience or test data. Equipment anchorage is analyzed for each component. Caveats and guidance are provided in the criteria screening tables in EPRI NP-6041 for three ranges of seismic input.
Required Experience & Training		
The seismic PRA should be performed by experienced systems and seismic capability engineers who can perform seismic fragility analysis.	The seismic margins assessment should be performed by trained, experienced seismic capability and systems engineers. Seismic capability engineers must be capable of performing fragility analysis if this method is used.	The seismic margins assessment should be performed by trained, experienced seismic capability and systems engineers.
Walkdown Procedures		
Principal elements of the walkdown are (1) seismic capacity versus seismic demand, (2) caveats based on earthquake experience and generic testing data bases, (3) anchorage adequacy, and (4) seismic-spacial interaction with nearby equipment, systems, and structures. Walkdown procedures should follow the requirements contained in EPRI NP-6041.	Principal elements of the walkdown are (1) seismic capacity versus seismic demand, (2) caveats based on earthquake experience and generic testing data bases, (3) anchorage adequacy, and (4) seismic-spacial interaction with nearby equipment, systems, and structures. Elements not screened out are identified as outliers for further review. Walkdown procedures should follow the requirements contained in EPRI NP-6041.	Same as USNRC seismic margins method.

Table 1. Comparison of Seismic PRA and Seismic Margins Methodologies (cont)

Evaluation of Component Capacity For elements not screened out during walkdown, calculate fragility parameter values, that is, median capacities and logarithmic standard deviations	The capacity of components that were not screened out can be calculated using the fragility analysis (FA) or the conservative deterministic failure margin (CDFM) method.	The capacity of components that were not screened out can be calculated using the conservative deterministic failure margin (CDFM) method

Table 2. Advantages and Disadvantages with Seismic Margins Methodology

Advantages	Disadvantages
Most important elements of seismic PRAs are retained: plant walkdowns and an ability to identify potential plant vulnerabilities through an integrated review of plant response.	No direct risk insights are obtained
The scope of components and systems that need reviews is reduced.	Accident mitigation, accident management, and emergency planning can be addressed only to a limited extent.
A measure of plant capacity is provided that is more easily understood and appreciated by engineers.	Nonseismic failures are addressed in an approximate manner by selecting success paths which are "highly likely to succeed".
Plant capacity estimates will be useful to judge the impact of design basis earthquake issues.	Ranking is based only on HCLPF capacities; thereby, making it difficult to prioritize issues in the absence of a better risk-based ranking.
Results are not affected by seismic hazard issues related to calculation of seismic return periods.	The system-screening guidelines as applied to a very old plant may require plant-specific modifications.
The level-of-effort required to implement is lower than that for a seismic PRA when both are done at the same level of detail.	It is more difficult to use when the perceived hazard is so high that the review level earthquake would be above the 0.3g and 0.5g (0.8g and 1.2g spectral acceleration) screening values.
Correlations among failures can be identified and analyzed with the USNRC event/fault tree method.	

Table 3. Advantages and Disadvantages with Seismic PRA Methodology

Advantages	Disadvantages
It can expand upon the event/fault trees developed for the internal events PRA analysis.	Correlations among failures can be identified and analyzed.
It provides a complete risk profile and can provide all the results obtained from the seismic margins methodology. Uncertainties are explicitly accounted for.	
Accident mitigation, accident management, and emergency planning can be addressed more systematically and with greater detail.	
Decision-making can be based on plant-specific risk results.	
Ranking based on different indices are available, for instance, core melt, frequency, release.	

The level-of-effort required is higher than the seismic margins methodology because of the enhanced scope when done at the same level.

Numerical results are often controversial because of large uncertainties and use of subjective judgement.

Because of the large uncertainties in the seismic hazard estimates, core damage frequencies and risk results are generally insensitive to changes in fragilities.

It may be inappropriately used to focus on bottom line numbers; thereby, introducing the tendency to make inappropriate comparisons with other initiators.

OECD-NEA Workshop on the Seismic Re-evaluation of all Nuclear Facilities,
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Seismic Re-evaluation of Kolzoduy NPP
Criteria, Methodology, Implementation

Dr. Marin Kostov
Risk Engineering Ltd, Sofia, Bulgaria

Introduction

Seismic design codes and standards for nuclear facilities are not available in Bulgaria. The existing facilities were designed either without taking into account any seismic loads (Units 1, 2 and partially 3, 4) or according the ex-USSR codes and standards (Units 5 and 6). After the 1977 Vrancea (Romania) earthquake most of the seismic safety codes for conventional construction as well as the design codes for nuclear facilities in Eastern Europe were reconsidered. The program for increasing of the seismic safety of the NPP Kozloduy started at that time. Peak ground acceleration 0.1g for the site of the NPP was estimated. Partial upgrading of the civil structures was performed and considerable measures for improving the seismic safety of the equipment were implemented – snubbers were installed for the primary circuit of Units 1 and 2. The snubbers for Units 3 and 4 were installed during the construction as "original design".

The seismic safety of Units 5 and 6 is significantly higher. The seismic analyses are performed according to the requirements of ПНАЭ Г-5-006-87 in force since 1988. The civil structures are designed for peak ground acceleration of 0.1g.

The paper is presenting some important elements of the procedure for seismic re-evaluation of the units of Kozloduy NPP. The analyses and the seismic upgrading of the building structures are discussed.

Methodology and Criteria for Re-evaluation

The new program for seismic re-evaluation of the Kozloduy NPP started in 1990 with a full-scope site conformation project. The projects finished in 1992 and the ground acceleration 0.2g was established as PGA of the review level earthquake. In the same year the six-month WANO program for increasing of the safety of Units 1 and 2 began. In the frame of this program an international expert team from IAEA drew up the Terms of Reference and Technical Specifications for Seismic Upgrading Design of Kozloduy NPP Units 1 and 2 (TOR). This document was the basis for the seismic re-evaluations of Units 1 to 4.

The modernization programs of Units 5 and 6 started at present as well as the plans for future investments implied the necessity of development of general rules for seismic re-evaluation and design of nuclear facilities in Bulgaria [3]. These rules should be applicable to the Radioactive Waste Storage Facilities too, taking into account the specific requirements. In 2000 a specific guideline for seismic re-evaluation (and design) of nuclear facilities was created within a project reviewed by the IAEA. The guideline is based on the documents of IAEA, codes of EC, codes of USA, Germany, Japan and other countries with developed nuclear technology. The list of codes and standards used in these guidelines comprises more than 150 items.

The guideline is aimed to help Bulgarian designer to understand and correctly apply the international technical documents. That is way most of the applicable code and standards are referred in the document. The document is not aimed to replace the original codes and standards. Emphasis is given to the proper selection of the re-evaluation (or design) philosophy and the design codes reflecting it.

The main goals of the guideline are:

- to summarize the Bulgarian experience over 10 years of intensive seismic upgrading of all 6 units of Kozloduy.
- to create basis for understanding of the different teams working on seismic projects in Bulgaria
- to create uniform basis for performing seismic upgrading in Bulgaria
- to help NPP-K in assessing the actual status and to plan the future actions needed

The guideline consists of the following parts:

Part 1. Main principles
Part 2. Geological and seismological requirements
Part 3. Seismic safety of civil structures
Part 4. Seismic safety of piping and equipment
Part 5. Seismic design and qualification of electrical and I&C equipment
Part 6. Seismic qualification by dynamic test
Part 7. Seismic monitoring and control

The Review Level Earthquake

The NPP Kozloduy is located in a region influenced by different seismic sources – far field sources of intermediate depth, shallow far filed and shallow near-field ones. The seismic-tectonic studies of the zone 320 km around the NPP site defined the characteristics of the review level earthquake (RLE). Those are: the maximum acceleration equal to 0.2g and 0.1g for horizontal and vertical components respectively (annual probability of exceedance 10^{-4}), and the free field broad band enveloped acceleration response spectra. The maximum spectral accelerations are constant within the period range from T=0.1s to T=1.7s. The frequency content of the spectra is one and the same for the horizontal and the vertical component. The RLE spectrum is shown in Fig. 1 and Fig. 2. The time histories generated to be compatible with the free field spectrum have a total duration of 60 seconds and intensive part 15 seconds.

The seismic review of NPP Kozoduy is performed also for effects of seismic excitations from local earthquakes. A zone with a radius of 30 km around the NPP site was subject of the investigation [2]. A seismic hazard analysis is carried out using different alternatives for variation parameters such as attenuation law, maximum and minimum expected magnitudes, magnitude-frequency relation, spatial distribution of the seismicity in the local zone. As a result the mean maximum acceleration due to the local seismicity is assessed to be 0.128g and the respective 85% confidence value is 0.162g for seismic level with annual probability of exceedance 10^{-4}. Those values refer to the horizontal components of the free field ground motion. The acceleration response spectra on the free field of NPP site for local earthquakes are established on the basis of acceleration time histories recorded at small epicentral distance. The vertical response spectrum due to the local earthquakes is larger than the RLE spectrum and the higher frequency character of the near-field ground motion is clearly expressed. The artificial accelerograms generated for local earthquakes have total duration of 20s and intensive phase of 2s. Their acceleration response spectra for 5% damping are shown also in Fig. 1 (SINTH1, SINTH2) and Fig. 2 (SINTV).

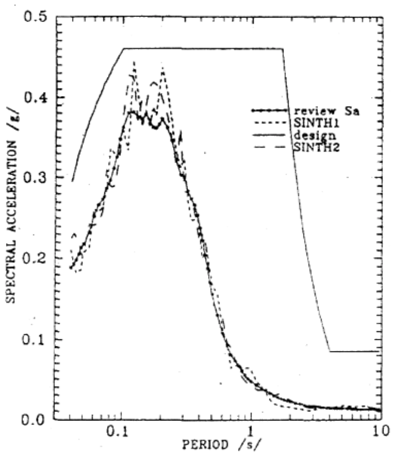

Fig1. Free field acceleration response spectra - horizontal componen

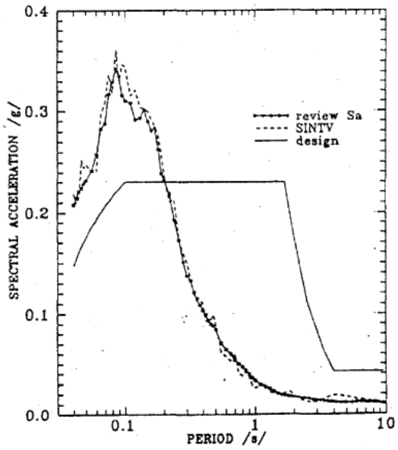

Fig2. Free field acceleration response spectra - vertical component

Damping and inelastic absorption factors

Behavior and safety of the structures subjected to seismic actions depend not only on the resistance and stiffness of the structural members but also on parameters like ductility and damping. The bigger those parameters are the bigger is the reduction of the effective seismic loading. The use of realistic damping factors and member ductility is one important feature of the seismic re-evaluation program.

To provide an appropriate ductile behavior of a reinforced structure it is necessary to ensure:

(1) Good ductility at curvature in all critical zones;

(2) Over-strength in the critical zones of the members, loaded in compression (columns) or shear (short beams or columns);

(3) Reinforcement with high plastic deformability;

(4) Concrete with high strength capacity;

(5) Ratio of the tensile to yield strength of the reinforcement steel higher than 1.3.

For steel structures higher ductility factors are permitted for members subjected to flexure and secondary structure elements. The ductile behaviour of members subjected to flexure is due to the work (energy) invoked by limited plastic deformations. Special zones for energy dissipation by plastification of the beam cross sections up to forming plastic hinges could be designed. Usually, these zones are arranged in beams near by their connections to the main vertical bearing elements.

The steel structures usually have a good ductile behavior and the brittle failure is not typical. However to prevent brittle failure it is necessary to limit the stress concentration, to avoid high shear combined with high tension, not to utilize the material hardening etc. It is recommended to use steel materials with pronounced yielding plateaux.

Table 1 presents damping coefficient values, required by linear elastic analysis, as a function of the Demand/Capacity for two response levels [5]. Demand is evaluated by elastic analysis, while Capacity is assumed as specified by codes. Response level 2 corresponds to D/C rations higher than 0.5. For generation of the floor response spectra the structure's response may be obtained by damping coefficients for response level 1 (level of stress condition 1). Damping coefficients for response level 1 should be used if the designed solution is controlled by stability [5]. To assess the existing structures and equipment, damping coefficients for response level 2 may be used in elastic analysis apart from the real stress condition reached.

Table 1. Specified damping values

Type of Component	Damping (% of critical) Response Level 1	Damping (% of critical) Response Level 2
Welded and friction bolted metal structures	2	4
Bearing-bolted metal structures	4	7
Prestressed concrete structures (without complete loss of prestress)	2	5
Reinforced concrete structures	4	7
Masonry reinforced shear walls	4	7
Wood structures with nailed joints	5	10
Distribution systems *	3	5
Massive, low stressed components (pumps, motors, etc.)	2	3
Light welded instrument racks	2	3
Electrical cabinets and other equipment	3	4
Liquid containing metal tanks		
Impulsive mode	2	3
Sloshing mode	0.5	0.5

* *Cable trays more than one half full of loose cables may use 10% of critical damping.*

The coefficient Fμ is used for reduction of the seismic loading or force, determined at elastic response of a structure. The reduction factor applied as correction of the internal forces obtained by linear analysis depends basically on the provided ductility of the structures. It may take into account also other factors influencing the real values of the effects. Such factors are: (1) the type of the structure (frame, wall, dual or combined) with its specific stress state; (2) the structural material; (3) the arrangement of the structure and the level of irregularity; (4) physical parameters of the connections, contribution of non-structural elements, etc.

The design internal forces due to seismic excitation shall be determined by dividing the internal forces obtained by linear analysis by the reduction factor Fμ given in Table 2. With insignificant correction the specified values of the reduction factors are corresponding to the values given in IAEA RU-5869 [4] and DOE-STD-1020 [5].

Table 2 Reduction Factors Fμ

Structural System	Fμ
FRAME STRUCTURES	
Reinforced concrete:	
Columns in flexure and axial compression	1.25
Columns in shear and/or predominant compression	1.00
Beams in flexure and axial compression	1.50
Beams in shear	1.25
Connections	1.00
Steel:	
Columns in flexure and axial compression	1.25
Columns in shear	1.00
Beams in flexure and axial compression	1.50
Beams in shear	1.25
Connections	1.00

Table 2 Reduction Factors Fμ(continued)

SHEAR WALLS	
Reinforced shear walls:	
In-plane bending	1.50
In-plane shear	1.25
Out-of-plane bending	1.50
Out-of-plane shear	1.00
Reinforced masonry	1.00
BRACED FRAMES	
Reinforced concrete:	
Columns in flexure and axial compression	1.25
Columns in shear and/or predominant compression	1.00
Beams in flexure and axial compression	1.50
Beams in shear	1.00
Bracing (Steel) :	
a) tensile	1.25
б) compression	1.00
Connections	1.00
Steel:	
Columns in flexure and axial compression	1.25
Columns in shear	1.00
Beams in flexure and axial compression	1.75
Beams in shear	1.50
Bracing (Steel) :	
a) tensile	1.25
б) compression	1.00
Connections	1.00
METAL LIQUID STORAGE TANKS	
Moment and Shear Capacity	1.25
Hoop Capacity	1.50

Notes:

Fμ refers to reassessment of existing nuclear facilities

Analytical procedures

The analytical procedures used for the seismic re-evaluation program should provide possibilities for realistic and non-conservative solutions. Usually time history analyses are used both for in-structure spectra generation and capacity analyses.

The analyses chain should include realistic definition of the seismic excitation at foundation level, i.e. deconvolution of the seismic free field motion is usually performed. The soil characteristics as well as the strain compatible soil properties are established experimentally. The equivalent linear method is frequently used to account for the soil nonlinear properties. The acceleration response spectra of the transferred to foundation level acceleration time histories (both RLE and Local Earthquake) are shown in Fig. 3 and Fig. 4 (SINTxx indicate the spectra from LE).

Important feature of the analytical procedure is the use of spatial FE models. The VVER structures typically connect elements with very different stiffness and geometry. The rotational response usually is an important factor. That is the reason the 3D models to be preferred in the seismic re-evaluation studies. The VVER structures are quite complex, several structures of seismic category 1 and 2 are interconnected and create complicated spatial systems that should be analyzed together.

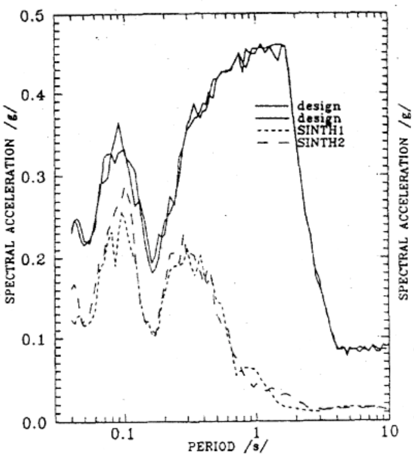

Fig.3. Deconvolved acceleration response spectra - horizontal componen

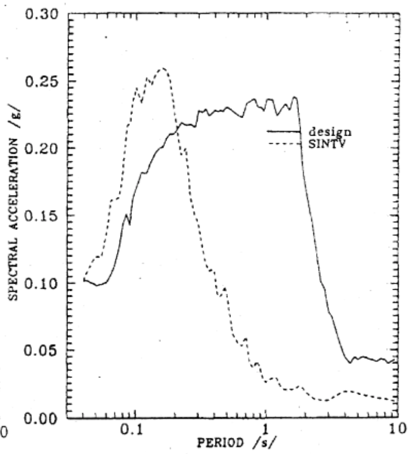

Fig.4. Deconvolved acceleration response spectra - vertical component

The analyses should take into account the soil-structure interaction. Especially for sites with relatively low sear wave velocity (at the Kozloduy site the average shear wave velocity of the surface layers is about 450m/s) the soil structure interaction may dominate the seismic response. The experience from the performed analyses shows that the soil-structure interaction is the most important factor for the response/capacity analyses of the VVER1000 units at the Kolzoduy site. Mostly the soil-structure interaction is analyzed by using the impedance analysis.

Strengthening techniques

The type of seismic strengthening of existing structures depends on the available seismic safety. Generally in case of small seismic capacity deficiency a local strengthening is performed. Otherwise in case of low seismic capacity a global strengthening should be accomplished.

Local Strengthening

Local strengthening of a structure is performed when the capacity deficiency is small, e.g., up to about 20%. Jacketing around an existing component is the most often used technique for local strengthening and increase of ductility of RC frame structures. Strengthening of steel structure components can be done by welding of additional steel plates or profiles.

Global Strengthening

Global strengthening using introduction of new structural subsystems can decrease the available irregularities in an existing structure and thus significantly improve its seismic behaviour. The new introduced subsystems should be as much as possible regularly distributed, without concentration in few places. By this manner will be avoided high values of forces in structural members and connections, e.g. high tensile forces due to overturning moments at the foundations. The most often used techniques include: (a) introduction of shotcrete, reinforced concrete or steel elements in the empty fields of skeleton structures or (b) introduction of complete new subsystems like reinforced concrete shear walls and cores, steel frames or trusses, external buttresses on new foundations.

As part of the global strengthening introduction of new or improved foundations is required for implementation of new shear walls, cores, frames, trusses, buttresses, etc., as well as for support of strengthened columns . Two principle solutions are possible: (a) deep beams over the existing footings, connected to the existing columns and thus making a common foundation structure, and (b) introduction of new foundation body incorporating the existing foundations. In the first case usually no excavation works are required; in case of sufficient space and bigger demand, the new deep beams can unify more than two foundations. In the second case, excavation works are required but foundation body with larger support area can be implemented. In case of weak soil and big loading, micro-piles can be implemented. In case of non-connected single foundations, especially when the soil is weak, a system of tie-beam foundations should be designed to unify the existing single footings.

Implementation: Seismic Upgrading of Main Building, Unit 3 and 4

As an example of implementation of the methodology for seismic re-evaluation the analyses of units 3 and 4 are presented below. At present the seismic analyses of all six units of Kozloduy NPP are already performed. The seismic upgrades of the auxiliary buildings as pump stations, diesel generators, spent fuel storage, etc. are completed. The upgrading of the main building of unit 3 and 4 is under implementation and will be finished till end of 2001.

The Units 3 and 4 of Kozloduy NPP are of type VVER 440-230. Both units are designed in the early 70's. They are twin structures. The main building of one unit consists of reactor building, turbine building, longitudinal electric building (electric spreading rooms), transverse electric building (motor control center, electric spreading rooms) and ventilation center. All those buildings are interconnected in one building complex. Unfortunately they have quite different dynamic behavior and the interfaces are very complex [1]. All buildings are RC frame structures. Only the reactor compartment is constructed of RC walls. The foundations of the frame structures are single footings. The reactor compartment is founded on a reinforced concrete mat. The foundation depth is approximately 7m below grade.

The two most important parts of the main building complex are the reactor building and the turbine hall. The longitudinal electric building connects these parts. Attached to the main structure are the ventilation center building and the transverse electrical building.

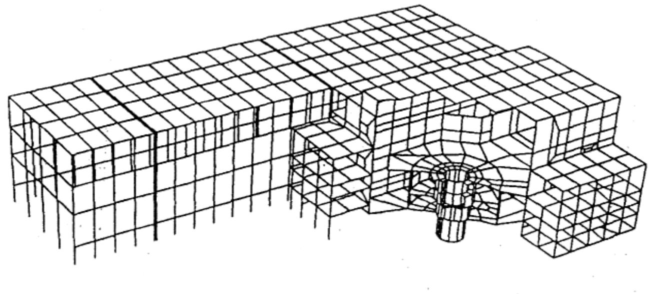

Fig.5. 3D FE Model, Unit 3/4

The turbine hall is monolithic reinforced concrete frame structure. It has 39m span. The roof consists of steel trusses covered by heavy RC panels.

The reactor building is also monolithic reinforced concrete structure. It consists of walls up to the elevation 10.5m (reactor compartment). Above that elevation the reactor building is frame structure with 39m span. The roof is similar to that of the turbine hall.

The longitudinal electric building connects the reactor building and the turbine hall. It is constructed on the columns of both buildings that are connected by prefabricated beams and panels (the floors of the electric building).

The detailed 3D finite element model is shown in fig.5. The model consists of 1345 beam elements and 2298 plate elements. The RC slabs and walls are modeled by plate elements.

The frame structures are modeled by beams. The outer wall panels (they are not bearing elements) are modeled only as lumped masses. Attention is paid to represent accurately in the model the actual releases of the prefabricated elements. The designed documentation is checked by walk-downs to reflect the as-built status of the structures.

The primary coolant system is modeled by beam elements and introduced in the structure model. This is done because of the significant weight of the reactor systems and the possible interaction with the building structure. The steam generators are attached to the structure by snubbers. Their dynamic behavior is also of interest.

The soil-structure interaction is modeled by spring-dashpot systems. The seismic excitation on free field is transferred to the foundation level by deconvolution.

The dynamic analysis is performed by the Stardyne 4.0 computer code. In Table 3 the first 10 eigen frequencies are presented.

Table 3. Eigen frequencies of the as-built structure model

Mode	Frequency	Period	Mode	Frequency	Period
1	1.46	0.68	6	2.81	0.35
2	2.37	0.42	7	2.92	0.34
3	2.54	0.39	8	3.17	0.31
4	2.58	0.38	9	3.25	0.307
5	2.69	0.37	10	3.33	0.30

The analysis of the mode shapes is showing that there is intensive rotation of all attached buildings (turbine hall, electric building and ventilation center) around the stiff reactor building. There is also intensive rotational motion in upper frame part of the reactor building.

The seismic effects are computed by time history analysis. The three components of the strong motion are applied simultaneously. The modal superposition technique is used. In the analysis 350 modes are computed. The activated modal mass is about 90 percent of the total mass. The material damping used is 0.07. The radiation damping is introduced for each footing. The modal damping is a composite one. For capacity analysis only the maximal computed effects are used.

The seismic effects (the member forces) are combined with the static one. The most unfavorable combination is aimed. In most of the cases these combinations are very conservative, e.g. maximum bending moments from static and dynamic analysis are added while the axial forces from the static analysis are reduced by the normal forces from dynamic excitation.

The capacity assessment is performed according to the Bulgarian standard for reinforced structures (limit state concept). The nonlinear reinforced concrete behavior is accounted by introducing member ductility. The typical member ductility for columns is 1.25 and for beams 1.5. No ductility is allowed for ties and connection. Generally all the outer column rows are overloaded. The deficiency in the stiffness is presented both in longitudinal and transversal direction. The computed displacements at several control points are bigger than the available gaps of the expansion joints, i.e dynamic impact between adjacent structure is possible. All these results are the reason for the proposed seismic upgrading.

Three different approaches are used for upgrading of the structure. The first approach is based on the local strengthening of elements with low capacity. It means in most of the cases to add more material, i.e. to increase cross section or inertia moment of the element. Usually this could be realized by jacketing of the existing elements or by introducing new elements acting with the existing ones simultaneously and in the same way, e.g. a parallel beam or column. Usually this approach is not very effective.

The second approach is based on the understanding of the global seismic behavior of the structure. By adding new elements it is aimed to improve the unfavorable response modes. Basically by adding new elements a better load path is achieved, the overloaded elements are unloaded and the not fully utilized elements are additionally loaded. This approach could be very effective.

The third approach is especially important for prefabricated structures. Usually they have RC elements of good quality. The loading capacity of those elements can not be used thoroughly because of the bad implementation of the element joints. The local improvement of the joints would contribute for better dynamic behavior of the whole structure.

A general view of the model of the upgraded structure is presented in fig.6.

The most unfavorable mode of seismic response for the main structure is the rotation of all the attached buildings along vertical axis going through the virtual center of stiffness, which is located in the reactor building. Because of this predominant feature of the response the most important upgrading are the new diagonal braces in the column rows placed at the periphery of the structure. Although these upgrading could not change the character of the response they contribute very much for improving the seismic safety of the frame structures. The second major upgrading is the roof connection between turbine hall and reactor building.

The aim of these new elements is also to help resisting the rotation response of the structure, to create continuity in the roof disk, to reduce the horizontal shear force in the columns of reactor building due to response of turbine hall. Improving the joints of the prefabricated elements of the longitudinal electric building increases the effect of these upgrading. Detail of the upgrading is shown in fig.6. The improvement (upgrading) of the joints is proposed also for the transversal electric building and for the roof-to-column connections in the reactor building and turbine hall. The first 10 eigen frequencies of the upgraded structure model are presented in table 4.

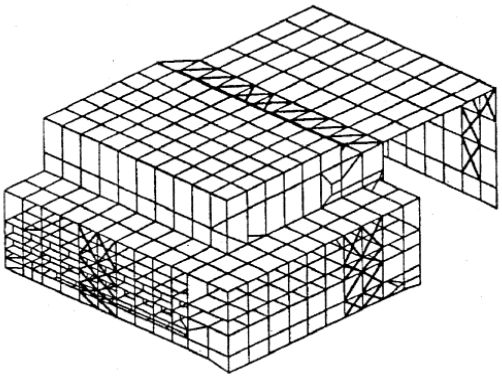

Fig. 6.Upgraded Structure model

The first natural frequency is increased from 1.46Hz to 2.1Hz. This is the only significant shift in the frequencies. Although the seismic upgrading is very limited (it is about 300t steel) the effect on the seismic capacity is significant. After the implementation of the upgrades there is no deficiency in capacity or in structure deflections.

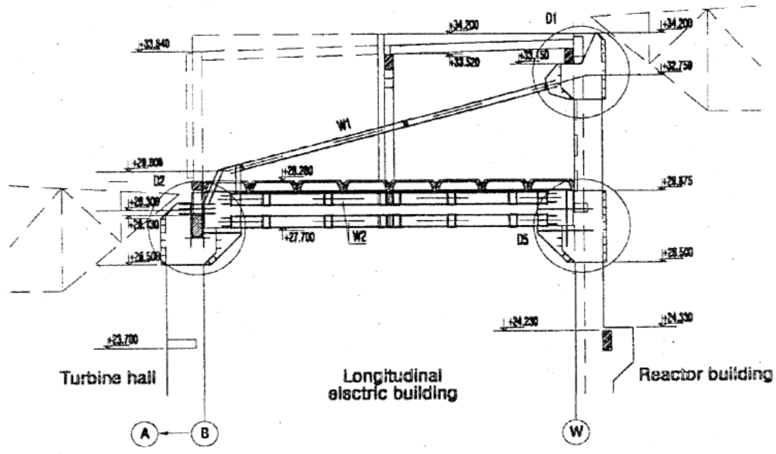

Fig. 6. Detail: Upgrading of roof between turbine hall and reactor building

Table 4. Eigen frequencies of the upgraded structure model

Mode	Frequency	Period	Mode	Frequency	Period
1	2.1	0.47	6	2.91	0.34
2	2.43	0.41	7	3.16	0.31
3	2.56	0.39	8	3.23	0.309
4	2.66	0.37	9	3.33	0.30
5	2.88	0.35	10	3.39	0.29

Conclusins

The paper describes some features of the methodology applied for seismic upgrading of civil structures at the site of NPP Kozloduy. The essence of the methodology is the use of as-build data, realistic damping and inelastic reduction factors.

As an example of seismic upgrading the analyses of unit 3 and 4 are presented. The analyses are showing that for effective seismic upgrading detailed investigations are needed in order to understand the significant response modes of the structures. In the presented case this is the rotation of the attached flexible structures to the stiff reactor building. Based on this an upgrading approach is applied to increase the seismic resistance for the predominant motion. The second significant approach applied is the strengthening of the prefabricated element joints. Although it is very simple it allows use of the available element capacity.

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Seismic Re-evaluation in the UK - A Regulator's Perspective

John Donald - HSE (United Kingdom)

Abstract

The seismic re-evaluation process in the UK has been ongoing for a period of approximately six years. For most of the plants considered to date, this has been an initial seismic evaluation as the plant had no consideration of seismic loading in the original design.

As the UK gas cooled reactors are diverse in terms of original design, the approach used has mainly taken the form of a deterministic assessment to establish a robust line of defence for trip, shutdown and decay heat removal for an infrequent seismic event at 10^{-4} per annum. Where possible, this has been supplemented by a second line of defence for a more frequent event, typically of the order of 10^{-3} per annum. Where the deterministic assessment has had difficulty demonstrating a fully acceptable case, then a seismic margins approach has been adopted in some situations. No power station has used a seismic PSA approach in full to date, but seismic risk calculations have been used to inform some ALARP (as low as is reasonably practicable) decisions regarding plant modifications. As the first round of periodic safety reviews comes to a close and a second round starts, it is possible that a seismic PSA approach may be adopted to demonstrate the ongoing validity of the seismic qualification completed to date, and that the risk is ALARP.

The SQUG (seismic qualification utility group) methods have been used for the seismic re-evaluation of equipment as part of the deterministic assessments. The SQUG approach has been supplemented by other more conventional analysis and code assessment methods for some piping and other equipment outside the SQUG classes of equipment.

For novel analyses and assessments an independent peer review process has been completed prior to submission of the completed safety case to the regulator. Limited peer review has been undertaken for more mundane work, and it has typically been completed on initial strategy documents and not during the re-evaluation main stream of work.

For the deterministic approach outlined above, a single success path has been re-evaluated against the specified seismic loading. Redundant or diverse safety systems have generally been considered for the more frequent earthquake evaluation.

In most cases, the criteria chosen to demonstrate success have been the same between re-evaluation and original design. This has not always been possible because of the differences in design detailing requirements, particularly for reinforced concrete structures, for example. The acceptance criteria have sometimes been amended when the reasonable practicability of the strengthening or countermeasures has been considered.

Introduction & Background

The inventory of power reactors within the UK includes a wide variety of type and forms of reactor plant. The majority of the power reactors are gas cooled, with a single PWR station at Sizewell. All of the early gas cooled reactors were designed and built with no seismic design basis. The more recent Advanced Gas cooled Reactors (AGR) and the PWR were designed to resist seismic loading amongst the many other design load conditions.

The Nuclear Installations Inspectorate regulates all of the UK power reactors. In addition, it also regulates the research reactors and other nuclear research facilities as well as fuel manufacturing and reprocessing facilities. The diversity of the facilities which NII regulates has led it to a goal setting regime, supported by a licensing system for all nuclear facilities. The law which underpins the regulatory regime, the Nuclear Installations Act and the Health and Safety at Work Act, also enshrine the concept of a goal setting regulator. The lack of prescription from the regulator is intended to allow the licensees to develop suitable methods which they fully understand and adopt because they are ‘owned’ by the licensees themselves.

The Periodic Safety Review (PSR) process, on a ten yearly cycle, is firmly established in the UK and many of the nuclear facilities have already completed the full PSR process, with further facilities currently undertaking PSR’s and a few yet to start, for example the Sizewell PWR which has not yet reached ten years of operation. Part of the PSR considered the ability of the facility to resist hazard loadings, both those included within the original design and loadings which were not part of the original design basis and safety case. The PSR has therefore provided the impetus for the facility managers and the regulators to consider seismic loading. As noted above, in many cases within the UK what has resulted is not a seismic re-evaluation but an initial seismic evaluation of an existing facility. Regardless of the terminology applied, the processes used in the initial seismic evaluation of a facility which had no seismic design basis are essentially identical to those used in seismic re-evaluation. This was confirmed by the use of identical procedures for AGR stations which previously had some seismic design basis and AGR stations which had no initial seismic design.

Definition and scope of the plant to be evaluated

Most nuclear power plants and facilities have a wide variety of safety systems available to the operators under fault conditions. For power reactors the safety systems must provide a number of essential functions including tripping the reactor, shutting the reactor down, providing holddown and providing a means of removing decay heat. Further to this a number of other functions may provide defence in depth, such as providing a containment which is sufficiently robust to withstand hazards loadings and accident conditions within the plant. However provided the essential safety systems perform their duties in a controlled manner, the containment may never be challenged.

As a wide variety of trip, shutdown, holddown and decay heat removal systems are available, the UK licensees have opted to seismically qualify only a subset of the available systems. This is a success path approach. The regulator recognises that the cost and difficulty of seismic qualification of all available safety systems makes full seismic qualification of all safety systems beyond reasonable practicability and thus has endorsed the success path approach. However, this endorsement has been tempered by a requirement that the licensees must demonstrate that systems and equipment within the safety systems which are not part of the success path must not impinge upon the seismic qualification. In simple physical terms, this could be a spatial interaction where a non-seismically qualified structure, system or component (SSC) falls on and damages a seismically qualified SSC. More complex interactions could include the inadvertent use of essential cooling water supplies by a non-qualified system due to valves changing state because of an undesired signal.

In general, the UK licensees have also adopted an approach which delivers a single line of protection to trip, shutdown, holddown and cooldown for an infrequent seismic event. This infrequent seismic event has been defined at the 10^{-4}/year event. A second line of protection has also been sought for a more frequent seismic event, typically selected as a 10^{-3}/year event. This second line of protection has been seismically re-evaluated on a reasonable practicability basis. In some cases, therefore, the licensee has presented to the regulator a fully seismically qualified second line of protection, in other cases, an argument has been presented that the costs of the re-evaluation of the second line of protection outweigh the benefits to be gained in terms of risk reduction and hence this second line of protection is not re-evaluated. The regulator in the UK thus has to consider the risks associated with a frequent seismic event with only one line of protection seismically qualified, compared with the costs (financial, time, dose burden to workers, etc.) of providing this qualification. The regulator thus has to judge not only the technical basis of the re-evaluation and any modifications required to achieve qualification of the SSC's against the seismic event, but also has to consider the suitability of the cost argument presented by the licensees.

Methods and acceptance criteria

The methods used in seismic re-evaluation vary depending upon a variety of criteria. These criteria include the following:

- The physical structure, system or component under consideration, including its inherent seismic capability.
- The availability of replacement SSC's or of diverse systems which can be put in place as safety systems during seismic re-evaluation.
- The costs associated with the method of re-evaluation chosen.
- The robustness of the safety justification required for the SSC.

In general, the regulator prefers the licensee to adopt current design standards and acceptance criteria for both new original design and for seismic re-evaluation. This is the simplest approach as all parties have a clearer understanding of what is required from a seismic safety case and its underlying design at any given point in time. However, the regulator also needs to recognise that in some cases and for some criteria, current design standards cannot be met and some reasonably practicable variation must be accepted.

A simple example of how current design criteria cannot be met would be the reinforcement details required in the seismic design of reinforced concrete elements. Evidence from real earthquakes over the past two or three decades has considerably revised the reinforcement detailing requirements. These changes have been implemented to make structures more damage tolerant and to ensure that system ductility demands can be accommodated. For an existing structure which was not designed using current seismic ductile detailing requirements, the acceptance criteria must be amended. In some cases, the re-evaluation will show sufficiently large margins that a weakness in detailing is not significant, in other cases, it can become the limiting feature (if all others are acceptable) and the licensee then needs to consider why the current design codes contain particular criteria and why a failure to meet them remains adequate - generally on a case by case basis.

Benefits and disadvantages of different methods (seismic PSA, Margins, deterministic, databases, tests, etc)

In the UK regulatory regime, there are two parallel thrusts to the criteria against which safety cases are judged for external events. These are deterministic criteria and risk based criteria. They are considered to be complementary.

For a new design, the UK regulator requires both the deterministic and the risk criteria to be met. For the re-evaluation of existing nuclear facilities, the current preference is for the deterministic criteria to be met in all cases and where reasonably practicable, for the risk based criteria to also be shown to be met. The implicit judgement in this strategy is that if a single, deterministically justified line of defence is qualified to resist an infrequent seismic event, then the risk should be tolerable. For facilities where two lines of protection are qualified for frequent events with one of these lines of protection qualified for an infrequent event, then the risk will be tolerable and may even lie in the lower-risk broadly acceptable region.

In the UK the confidence in deterministic qualification from the regulator is matched by a similar confidence from the licensees. At present, few of the UK facilities which have been the subject of seismic re-evaluation have had any form of seismic PSA or other means of risk calculation applied. However, as the licensees are moving towards consideration of the reasonable practicability of some modifications, particularly for frequent seismic events, the need for seismic risk calculations is strengthening. Some modification processes are now being selected by consideration of seismic risk and re-evaluation methods based on a simple seismic PSA to inform ALARP (As Low As Reasonably Practicable) arguments.

The emphasis which the UK regulator has placed on the deterministic methods within the initial round of PSR was soundly based considering the lack of a robust seismic case for many of the facilities and the desire to quickly introduce such a case. As the second round of PSR's approaches and the original seismic justifications are reviewed, it may be that more emphasis will be placed on seismic risk calculations by the regulator. The use of seismic PSA may serve to confirm that the work completed in the original round of PSR's was soundly based, and can also inform how and where further modifications or strengthening should be applied to achieve the most cost effective safety benefit. The increasing use of seismic PSA based approaches throughout the world and to facilities other than PWR reactor plants is encouraging this view by the UK regulator.

The deterministic approaches which have been adopted by the UK licensees for the first round of PSR's have tended to follow patterns in terms of the methods adopted for different forms of SSC. For civil engineering structures and large safety components such as major water storage tanks, a deterministic method has generally been adopted. These methods have involved detailed finite element models of the structures supported by a design code based re-evaluation of the structure or component. The UK regulator has generally supported this type of approach for these types of structure and component.

For safety systems and individual safety system components a Seismic Qualification Utilities Group (SQUG) type approach has generally been adopted. These have been based on a process including safety system reviews, plant walkdowns and screening based seismic evaluations, all founded on seismic experience methods. The experience based methods are not considered as robust as deterministic design based assessment or seismic qualification by testing, a view shared by both the UK licensees and the regulator. The experience based methods are, however, a pragmatic approach for vintage equipment and for unique equipment which has few spares available for testing. The advantages of applying these methods is that it is straightforward and founded on evidence of past performance. The key to their usage lies in the skills and knowledge of the people applying the methods and the application of peer review and checking and verification to ensure that the results are controlled, consistent, structured and sufficiently robust.

For specific types of equipment seismic qualification by test has been the only viable means of establishing a robust seismic safety case. In almost all situations, the only equipment which has required this expensive form of seismic qualification has been replacement equipment, or individual relays. For the seismic re-evaluation of relays, some weight has been given to the information supplied from seismic experience data. Specifically some forms of relays are particularly vulnerable, some are particularly robust, and many lie between the two. Experience methods have therefore been able to define those relays which require complete replacement, and have also been used to identify those which are inherently reliable and for which a case can be made without testing. For all other relays, the seismic safety cases have required testing to confirm their seismic capability. This use of a combination of experience methods, replacement and testing has been endorsed by the UK regulator as a pragmatic means of establishing a seismic re-evaluation for relays.

Role and scope of peer review process

All of the methods adopted by a licensee for the purpose of seismic re-evaluation will be subject to some form of internal verification prior to their issue to the regulator for approval. In some cases, the verification can be extended to include external bodies and can be supplemented by a peer review process, again either internal or by independent external bodies.

Regardless of the method adopted by the licensee for seismic re-evaluation, the UK regulatory regime and the licensee's internal arrangements require a formal verification of the resulting safety case and of the underlying reports and other work. Depending upon the degree of innovation which has been involved in the process, a peer review process can strengthen the resulting safety case. For cases which are novel or which utilise existing methods, but perhaps with smaller margins than normally adopted, then a peer review process carries advantages for the licensees and the regulator, which can be demonstrated by the examples below:

In one case, a peer review was performed on an analysis-based seismic re-evaluation which involved a structure with a high degree of non-linearity and whose consequences of failure could be very severe. The high seismic reliability required of this structure (low fragility) and the significant extent of minor damage calculated to occur at low seismic input levels placed a high demand on the robustness of the safety case presented. In such circumstances, the use of an independent peer review of the work added to the confidence of the licensee submitting the case and of the regulator who assessed it.

A second area where peer review is commonly used is in the seismic experience methods of qualification. As noted previously, the application of this method places a high reliance on the personnel who carry out the walkdowns and effectively qualify significant quantities of equipment during the walkdown based on screening tables. Again, a peer review can strengthen any case presented because of the independent view of the personnel used and of their outputs. An experienced peer reviewer should be able to quickly identify a number of areas on the plant which may be weak and can then focus their review on the documentation which supports the seismic qualification of this system.

Countermeasures/strengthening

In many cases in the UK, the original review of the SSC's for each station has identified that an interim seismic safety case can be constructed and endorsed, but the full deterministic seismic safety case for both frequent and infrequent seismic events cannot be established without seismic strengthening. In some cases, as well as strengthening of the SSC's there has also been a need for the development of a series of instructions and documentation which define post seismic event actions required to be performed by the station operators. Such post-earthquake actions can be included within the term countermeasures. Countermeasures can also include other modifications which have been introduced to overcome a difficulty posed by failure of non-safety systems, but which did not require seismic qualification of the safety systems. A specific example is the use of battery backed emergency lighting. In areas where an operator has to perform a specific function after an earthquake, it cannot be guaranteed that there will be sufficient ambient lighting to allow him/her to perform that task. A simple countermeasure is to install a battery backed emergency lighting system which is not in itself safety related, but permits a safety related task to be performed.

In almost all cases, the strengthening which the UK licensees have had to undertake has been associated with improving the anchorage or structural integrity of SSC's. If there is a deficiency in the post seismic event functional performance of the equipment, it generally becomes very difficult to improve this performance by strengthening, and replacement of the SSC by a tested and deterministically qualified component has been required. The standards and acceptance criteria which have been applied to anchorage and structural integrity enhancements have typically been identical to those for new build. Although this has not been unconditionally required by the UK regulator, the licensees have adopted new build design standards for a number of reasons. Undoubtedly one of the reasons for this choice is the cost and difficulty associated with persuading the regulator that the acceptance criteria chosen will generate a sufficiently reliable outcome. The most important reason however, is the relatively low cost of the actual materials and components involved in the strengthening compared with the overall cost of the process. Most of the costs are associated with the installation of modifications, that is control, supervision, inspection, active area working etc, and the contribution from the materials themselves is usually low, hence any slight variation in the design usually has a very small effect on the overall cost.

Strategies and priorities

As the seismic re-evaluation process has been part of the Periodic Safety Review (PSR) process, it has been only one part of a multifaceted review of each facility and its safety case. For some hazards, the PSR has confirmed the adequacy of the existing safety case, but for many others some enhancement of the original case has been required.

The short timescales associated with some of the early PSR's carried out in the UK were such that it was not always possible for the licensees to implement all the countermeasures and strengthening required to provide a full deterministic seismic safety case for frequent and infrequent seismic events at the PSR due date. Under such circumstances both the licensees and the regulator had to agree what would be a suitable priority and to develop strategies to ultimately provide full seismic safety cases.

In many cases at the initial PSR deadline date a justification has been presented to the regulator of a interim safety case providing one line of protection for a infrequent, or sometimes frequent seismic event. This interim position has been supported by a strategy to achieve further improvements supported by a clear program of work to provide a full seismic safety case within a reasonable and realistic timescale.

Recent Research

In the UK nuclear industry, research is commissioned by the licensees alone, by the regulator alone or by all interested parties on a joint basis. The main strands of research associated with seismic re-evaluation have been on the following subjects:

- The use and validity of seismic PSA for gas cooled reactors. In this research, the extensive use of seismic PSA in the USA and in other countries has been reviewed to determine the most appropriate method which can be applied to gas cooled reactors. The research has also reviewed some of the unique SSC's associated with gas cooled reactors to determine if realistic seismic fragilities can be determined.
- The attenuation laws used in the calculation of seismic hazard at the various stations have been studied.
- The performance of infill masonry panels within heavy reinforced concrete frames has been studied both by analytical means and by scale testing on shake tables. This research has shown that whilst infill masonry panels perform quite poorly in the frames of conventional structures, their performance within heavy reinforced concrete structure typical of the UK nuclear industry is significantly better,

mainly due to the enhanced restraint from the surrounding framework which allows arching action to develop.

Conclusion

The seismic re-evaluation process has been successfully applied in the UK for a period in excess of six years. The process has formed part of the overall Periodic Safety Review for nuclear facilities. The regulation of the seismic re-evaluation process is similar to the regulation of the many other diverse activities performed by the licensees, but has some specific technical issues which require a more detailed view.

BENEFITS OF CHOOSING THE GIP METHODOLOGY FOR SEISMIC VERIFICATION OF EQUIPMENT.

THE SANTA MARÍA DE GAROÑA NUCLEAR POWER PLANT EXPERIENCE.

Dora Llanos.
NUCLENOR, S.A.
Santander. Spain.
dora.llanos@nuclenor.es

INTRODUCTION

THE SANTA MARÍA DE GAROÑA NUCLEAR POWER PLANT

The Santa María de Garoña Nuclear Power Plant (SMG NPP) is a BWR-3 owned by NUCLENOR, S.A. It has 466 MW electric power, General Electric technology, Mark-I primary containment and was designed by GETSCO-EBASCO. The plant started its commercial operation in 1971.

The original seismic design of the plant was made according to the seismic criteria in those years, i.e. using the Uniform Building Code for structures and static load coefficients for equipment. This original seismic design has been improved during the life of the plant by means of some very extensive and deep programmes; the most important ones are the Systematic Evaluation Program, that affected to structures and pipes, and the Seismic Verification of Equipment Program, that covered mechanical and electrical equipment.

GIP METHODOLOGY BACKGROUND

The methods for the seismic design of nuclear power plants have evolved since 1970 from the use of static load coefficient approach to more sophisticated methods today. Current nuclear seismic design methods for new plants consist of detailed dynamic analysis or testing of safety related structures, equipment, instrumentation, controls, and the associated distribution systems (piping, cable trays, conduit and ducts).

Because of the amount of changes in the design requirements, the NRC started USI A-46, “Seismic Qualification of Equipment in Operating Plants”, in December 1980, to address the concern that older operating nuclear power plants contained equipment which may not have been qualified to meet the newer, more rigorous seismic design criteria. It was realised that it would not be practical and cost-effective to do the seismic qualification of safety related equipment using procedures applicable to plants under construction. The objective of USI A-46 was to develop alternative methods and acceptance criteria which could be used to verify the seismic adequacy of essential mechanical and electrical equipment in operating nuclear power plants.

In 1982, the Seismic Qualification Utility Group (SQUG), was formed for the purpose of collecting seismic experience data as a cost-effective way of verifying the seismic adequacy of equipment in operating plants. The sources of experience data are equipment in non-nuclear power plants and industrial facilities which have experienced significant earthquakes, and shake table tests data used to qualify safety related equipment for licensing of new nuclear plants. SQUG has organised this information and has developed guidelines and criteria for its use; they are contained in the Generic Implementation Procedure (GIP).

SEISMIC VERIFICATION OF EQUIPMENT PROGRAM IN THE SMG NPP

In the late 1980’s, NUCLENOR joined the Seismic Qualification Utility Group (SQUG) and began to outline the Seismic Verification of Equipment Program, following the GIP guidelines and criteria. The Spanish Regulatory Body asked NUCLENOR to perform the Program in 1993.

OVERVIEW OF THE GIP METHODOLOGY

The steps followed in the GIP methodology are:

Selection of seismic evaluation personnel. The Seismic Capability Engineers (SCE) perform the screening verification and walkdown of the safe shutdown equipment. Since the use of GIP guidelines require engineering judgement because they are not inflexible rules, the SCE must have sound seismic engineering knowledge and experience and have to complete an SQUG-developed training course.

Identification of safe shutdown equipment. The first step is to define two paths (preferred and alternative) which could be used to accomplish the functions needed to shut down the plant after an earthquake. The second step is to identify the active mechanical and electrical equipment, tanks, heat exchangers, and cable and conduit raceway systems in these paths. Screening guidelines are provided for evaluating the seismic adequacy of most of this equipment.

Screening verification and walkdown. The purpose of the screening verification and walkdown is to screen out those equipment which passes some generic seismic adequacy criteria. This screening is based on the use of the seismic experience data and you have to consider four main areas:

- Comparison of the equipment seismic capacity to the seismic demand imposed.
- Determination that the seismic experience data is applicable to the specific equipment (caveat).
- Evaluation of the equipment anchorage seismic adequacy.
- Check for adverse seismic spatial interactions.

The equipment which do not comply with all of the checks above are considered “outliers”.
In order to make the considerations above easier, equipment are grouped in twenty classes; they are: motor control centers, low voltage switchgear, medium voltage switchgear, transformers, horizontal pumps, vertical pumps, fluid-operated valves, motor-operated valves and solenoid operated valves, fans, air handlers, chillers, air compressors, motor-generators, distribution panels, batteries on racks, battery chargers and inverters, engine-generators, instruments on racks, temperature sensors and instrumentation and control panels and cabinets.

Outlier identification and resolution. Outliers are identified considering the four areas above. Methods of outlier resolution are more time consuming and expensive than the screening evaluations and depend on the screening criteria not met and by how much, the need of analytical evaluation, the number of times the outlier is repeated, etc.

Relays, tanks and heat exchangers, and cable and conduit raceway systems are not included in the previous screening criteria and they have to be evaluated separately.

APPLICATION TO SANTA MARÍA DE GAROÑA NPP

Seismic evaluation personnel. The project was directed by a NUCLENOR structural engineer with more than twenty years experience in seismic matters; the Seismic Capability Engineers were four freelance structural engineers with more than five years' seismic experience. They all attended the SQUG-developed training course.

Safe shutdown equipment list. The safe shutdown equipment list (SSEL) was developed by a NUCLENOR system engineer who attended the specific SQUG training course. The consequent list included four hundred and fifteen pieces of equipment and the number of units in each of the twenty classes are summarised in Table I.

Screening verification and walkdown. A very important effort was made in collecting data about the equipment (specifications, drawings, calculations, etc.) previous to the plant walkdowns. Both tasks were necessary to analyse the four hundred and fifteen items, case by case and step by step. The most important points to consider in each step are the following:

> *Capacity vs. demand*. The capacity is given by the Bounding Spectrum (BS. Figure 1), Generic Equipment Ruggedness Spectrum (GERS) or documentation (DOC) of equipment-specific seismic qualification data. The Bounding Spectrum is based on earthquake experience data and represents the lower envelope of the earthquake´s spectra considered to build the data base; there is only one Bounding Spectrum for the twenty classes. GERS are based on generic seismic test data and establish a different generic ruggedness level for some specific classes. Documentation is based in specific seismic qualification tests or in data on similar equipment.
>
> Depending on the location above the effective grade, the demand is given by the SSE Ground Response Spectrum (GRS) or the In-Structure SSE Response Spectrum (IRS).
>
> The comparison between these spectra may be affected by different coefficients depending on special circumstances.
>
> The SMG NPP SSE Ground Response Spectrum is the RG 1.60 spectral shape, 5% damping, anchored to 0.1g (GRS. Figure 1). This spectrum was used to develop all the In-Structure Response Spectra for the plant.

The number of SSEL units verified with each class of spectra is:

Capacity Spectra		Demand Spectra	
BS	391	GRS	331
GERS	9	IRS	84
DOC	15		

Caveats. Caveats are a set of inclusion and exclusion rules which represent specific characteristics that are very important for the seismic adequacy of a particular class of equipment. To apply caveats to an item of mechanical or electrical equipment, you should confirm that the equipment characteristics are generally similar to the earthquake experience equipment class or to the generic seismic testing equipment class. Caveats are not inflexible rules, so that engineering judgement should be used to determine whether the specific seismic concern addressed by the caveats is met. Caveats are generally met by all the equipment classes in SMG NPP.

Anchorage. This is probably the most important aspect in the verification, because the lack of anchorage or inadequate anchorage has been a significant cause of equipment failing to function properly during and following past earthquakes. The verification of seismic adequacy of equipment anchorage is based on a combination of inspections, analyses, and engineering judgement. 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 with demand. The typical type of anchorage in SMG NPP (expansion anchor bolts, embedded anchor bolts and welds) is covered by GIP and generally anchorage requirements are met. Only in thirty four units the seismic capacity was not greater than the seismic demand.

Seismic interaction. The final point in the verification of the seismic adequacy of equipment is to confirm that there are no adverse seismic spatial interactions with nearby equipment, systems, and structures which could cause the equipment to fail to perform its intended safe shutdown function. The concerns are proximity effects, structural failure and falling, and the flexibility of attached lines and cables. In SMG NPP, seismic interactions were found not to be a special concern. Only two types of seismic interactions were found: impact of adjacent cabinets containing essential relays and control room ceiling. The number of units affected is apparently high (forty six), but most of them are caused by the control room ceiling.

Outliers. An outlier is an item of equipment which does not comply with all the screening guidelines provided in the GIP. However, if an item of equipment fails to pass these generic screens, it may still be shown to be adequate for seismic loading by additional evaluations. The outliers found in SMG NPP are summarised in Table I. In this Table you can see the number of SSEL units that are considered outliers and which of the steps above these units do not comply with.

Table I. Number of equipment and outliers in SMG NPP

Equipment Class	Total number of items	Capacity vs. Demand	Caveats	Anchorage	Interactions
#1. Motor Control Centers	10	-	3	7	2
#2. Low Voltage Switchgear	2	-	2	-	-
#3. Medium Voltage Switchgear	2	-	2	1	-
#4. Transformers	2	-	-	-	-
#5. Horizontal Pumps	4	-	-	-	2
#6. Vertical Pumps	10	-	-	4	4
#7. Fluid-operated Valves	265	-	-	-	4
#8A. Motor Operated Valves	26	-	4	-	1
#8B. Solenoid-operated Valves	8	-	-	-	-
#9. Fans	6	-	-	-	-
#13. Motor Generators	2	-	2	-	-
#14. Distribution Panels	12	-	-	4	4
#15. Batteries on Racks	3	-	-	3	-

Equipment Class	Total number of items	Capacity vs. Demand	Caveats	Anchorage	Interactions
#16. Battery Chargers and Inverters	6	-	2	3	-
#17. Engine Generators	2	-	2	-	-
#18. Instruments on Racks	11	-	-	2	1
#19. Temperature Sensors	16	-	-	-	-
#20. Instrumentation and Control Panels and Cabinets	28	-	4	10	28
TOTAL	415	0	21	34	46

Relays, tanks and heat exchangers, and cable and conduit raceway systems are not included in the previous screening criteria and they have to be evaluated separately.

Relays. In the GIP context, a relay is a device with electrical contacts which change state and are vulnerable to chatter during a seismic event. It is only necessary to verify the seismic adequacy of relays (essential relays) whose malfunction precludes taking the plant from a normal operating condition to a safe condition. The term “malfunction” includes chatter of the contacts in the relay itself and any other spurious signals from other devices which control the operation of the relay. There are three methods that can be used to establish the seismic capacity of essential relays: generic seismic tests data, earthquake experience data, and relay-specific test data. There are four screening methods for comparing the seismic capacity of an essential relay to the seismic demand imposed upon it. If none of these screening methods result in an acceptable comparison of seismic capacity with demand, then the relay should be classified as an outlier. If an essential relay fails this generic screen, it may not necessarily be deficient for seismic loading; however, additional evaluations are needed to show that it is adequate. In SMG NPP there are two hundred and seventy seven (277) essential relays and one hundred and forty four (144) of them were found to be outliers; eighty six (86) relays were outliers because the capacity was not greater than demand and fifty eight (58) relays were outliers because of the lack of documentation about its seismic capacity.

Tanks and heat exchangers. It is necessary to perform an engineering evaluation on vertical and horizontal tanks and heat exchangers to check the seismic adequacy of: tank wall stability to prevent buckling (including the effects of hydrodynamic loadings) and tank wall flexibility; anchor bolt and embedment strength; anchorage connection strength between the anchor bolts and the shell of the tank or heat exchanger; and flexibility of piping attached to large, flat–bottom, vertical tanks. In SMG NPP there are nine tanks and the results are the following:

Equipment Class	Total number of items	Capacity vs. Demand	Caveats	Anchorage	Interactions
#21. Tanks and Heat Exchangers	9	-	1	4	2

Cable and conduit raceway systems. There are four steps to consider in cable and conduit raceway review: verify that the cable and conduit raceway systems meet the “Inclusion Rules”; evaluate the “Other Seismic Performance Concerns”; select a sample of representative worst–case raceway supports; and judge whether there are any seismic spatial interactions which could adversely affect their performance. In SMG NPP all 1E class cable raceway systems were inspected and bounding cases were calculated. The total number of raceway supports was eight hundred and fifty nine (859) and all of them were outliers because of the fail in the screw connections from the support to the raceway. Besides that, forty eight (48) of them failed in the frame supports and two hundred and seventeen (217) failed in the expansion anchor bolts.

OUTLIER RESOLUTION

As a result of the application of the screening verification and walkdown criteria, sixty three (63) outliers were found. In 1996, NUCLENOR developed a program to improve all these seismic-weak points that included: anchorage modifications in some electrical cabinets, control room panels, instrumentation racks and metal supporting structures; bolting of adjacent cabinets containing essential relays; control room ceiling details improvements; additional calculations for tanks and heat exchangers; change of relays; and raceway systems supports modifications. These tasks have been developed during these last five years and they shall be completed at the end of the ongoing outage.

OTHER BENEFITS

IPEEE

The main objective of the Individual Plant Examination of External Events (IPEEE) is a safety reevaluation in order to define the margin above the SSE that exists in operating plants. There are three approaches to conduct a seismic margin demonstration: a seismic probabilistic risk assessment (SPRA), a deterministic seismic margin assessment, and a combination of both.

Because of the similarity in methodology and mechanical and electrical equipment scope between the SQUG GIP and the Electric Power Research Institute (EPRI) “Conservative Deterministic Failure Margin Approach” (CDFM), NUCLENOR decided to apply the deterministic seismic margin assessment to take advantage of the large volume of work performed during the USI A-46 Program. For this reason the IPEEE program was conducted in a very cost–effective way.

NARE

The GIP was developed by SQUG primarily as a practical method for demonstrating the seismic adequacy of installed equipment in older, operating nuclear plants. However, recognising the value of the GIP approach for future procurement of new and replacement equipment and parts (NARE), provisions were included for application of the GIP to new and replacement equipment and raceways. NARE includes the new and/or replacement electrical and mechanical equipment, tanks, heat exchangers, relays, and electrical raceways.

Thanks of the Seismic Verification of Equipment Program, NUCLENOR acquired an important experience in the use of the GIP methodology. Now this experience is being used to verify the seismic adequacy of new and replacement equipment in a high quality, fast and cost-effective way.

In SMG NPP, the GIP methodology was used to verify the seismic adequacy in the following NARE cases: lubrication circuit modification in two emergency diesel generators, four motor control centers, twelve instrumentation and control panels and cabinets, four motor-operated valves, three batteries on racks, eight instrumentation racks, replacement of one horizontal pump motor, two chillers, six air handlers and six temperature switches.

CONCLUSIONS

As a result of the Seismic Verification of Equipment Program, all the seismic vulnerabilities of the plant were detected in a very cost-effective way. The number of equipment that did not pass the verification was very low in comparison with the total. The reasons that the equipment did not pass were related to a few causes and all of them were easily affordable. The most important effort was made in cable raceway systems supports improvements.

As a consequence of this effort, Santa María de Garoña Nuclear Power Plant has a level of seismic adequacy comparable to the level of newer plants and a very useful experience to deal with the seismic aspects of new projects.

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0.25	0.044
2.5	0.313
4	0.294
9	0.261
33	0.1
100	0.1
FRECUENCIA	GUÍA REG.

0.25	0.044
0.5	0.0794
0.75	0.1122
1	0.1434
1.25	0.1734
1.5	0.2026
1.75	0.2310
2	0.2589
2.25	0.2862
2.5	0.313
4	0.2928
5	0.2837
6	0.2764
7	0.2705
8	0.2654
9	0.261
11	0.2250
13	0.1989
15	0.1790
17	0.1632
19	0.1503
21	0.1396
23	0.1305
25	0.1227
27	0.1160
29	0.1100
31	0.1047
33	0.1
100	0.1

2	0.98	0.653
2.15	1.05	0.700
2.5	1.20	0.800
4	1.20	0.800
4.73	1.20	0.800
5	1.20	0.800
5.57	1.20	0.800
5.89	1.20	0.800
6	1.20	0.800
6.295	1.20	0.800
6.57	1.20	0.800
6.71	1.20	0.800
6.865	1.20	0.800
7.185	1.20	0.800
7.5	1.20	0.800
7.66	1.18	0.787
8	1.13	0.753
8.48	1.05	0.700
8.705	1.03	0.687
9.3	0.98	0.653
10	0.90	0.600
10.96	0.85	0.567
12	0.80	0.533
16	0.68	0.453
20	0.59	0.393
28	0.53	0.353
33	0.50	0.333
100	0.50	0.333
Frecuencias	1,5 x B.S.	BS

SEISMIC MARGIN ASSESSMENT OF SPANISH NUCLEAR POWER PLANTS: A PERSPECTIVE FROM INDUSTRY AND REGULATORS

Juan Garcia-Monge <g.monge.jm@cti.initec.es>
INITEC, Padilla, 17 - 28006 Madrid

Francisco Beltran <tbg@madrid.idom.es>
Ingenieria DOM Internacional, Jose Abascal; 4 - 28003 Madrid
and
Jose G. Sanchez-Cabanero <jsc@csn.es>
Consejo de Seguridad Nuclear. Justo Dorado, 11 - 28040 Madrid

INTRODUCTION

The worldwide experience with probabilistic safety analysis (PSA) of nuclear power plants shows that the risk derived from earthquakes can be a significant contributor to core damage frequency in some instances. As a consequence, no severe accident safety assessment can be considered complete without giving, due consideration to seismic risk. This fact has been recognized by some regulators. in particular, by the U.S. Nuclear Regulatory Commission (NRC), who has included seismic risk assessment in its severe accident policy (Ref. 1, 2).

The NRC's severe accident policy was adopted by the Spanish nuclear regulator. the Consejo de Seguridad Nuclear (CSN). As a result. all plants in Spain were asked to perform a seismic risk analysis according to Supplements No. 4 and 5 of Generic Letter 88-20 (Ref. 1) and NUREG-1407 (Ref. 2), which included the containment failure analysis. At present in Spain there arc nine operating reactors at seven sites: six Westinghouse-PWR, two GE-BWR and one Siemens/KW U-PWR. The vintages are very different: the oldest plant started commercial operation in 1968 and the most recent, in 1988.

In this framework, the Spanish Owners Group (SOG) proposed to CSN in 1994 to carry out the seismic risk analysis of the plants using seismic margin methodologies. This kind of methods requires, as a starting point, the definition of' a seismic margin earthquake (SNIE), also called review level earthquake (RLL). For this purpose, tile SOG sponsored a general Probabilistic Seismic Hazard Analysis (PSHA) for the seven Spanish sites. The results of this PSHA were used by the SOG to define tile RLE and the scope of the study for each plant (binning of plants). The proposal was submitted to the CSN for evaluation.

The CSN evaluation was based on the NRC practical experience and was helped by the technical advise of US Lawrence Livermore National Laboratory. The review showed that the uncertainties on seismic hazard had not been fullly captured and that it would have been justified to consider a more conservative binning of Spanish plant sites. Alter some lengthy discussions between the CSN and the utilities, the CSN staff accepted. as a reasonable minimum, the seismic hazard derived in the available PSI1A for each plant site.
Then, the CSN used this PSIIA to rank the plants in terms of hazard and assigned an SME and a scope for the seismic margin study at each plant (Ref. 3). Table 1 shows the final binning of the Spanish plants to be

used for SMA as an alternative option to seismic PRA. Note that all plants were assigned a RLE at the 0.3g level, like the vast majority of American plants to the East of the Rocky Mountains.

Site	Reactor type	**Power (Mwe)**	Operation since	SSE	Seismic categorization (NUREG-1407, Ref. 2)
Jose Cabrera	Westinghouse-PWR	160	1968	0.07 g	0 3 g, focused scope
Sta. Ma. de Garona	GE-BWR	466	1971	0.10 g	0.3 g, full scope
Almaraz I. ll	Westinghouse-PWR	974, 983	1981. 1983	0.10 g	0.3 g, focused scope
Ascó I, II	Westinghouse-PWR	1028 - 1024	1982, 1985	0.13 g	0.3 g, full scope
Cofrentes	GE-BWR	994	1984	0.17 g	0.32.g, full scope
Vandellós II	Westinghouse-PWR	1004	1988	0.20 g	0.3 g, full scope
Trillo	Siemens KWU-PWR	1066	1988	0.12 g	0. 3 g, focused scope

Table 1. Seismic categorization of Spanish sites for seismic margin assessments (SMA)

As a result of the previous process, and starting in 1996, the CSN has promoted a number of research projects in order to assess the uncertainties about seismic hazard at Spanish sites. The total budget for this projects is around 1.3 million EUR, and the results will be included in a future PSHA with expert opinion, following the guidelines in NUREG/CR-6372 (Ref. 4).

At the beginning of year 2001, all Spanish plants have completed their seismic assessments and most of these assessments are being reviewed by the CSN. The purpose of this paper is to give a wide perspective of the seismic margin studies performed in Spain. The paper is organized into three main sections. The first section is devoted to the implementation of seismic margin methodologies for Spanish plants. The focus is on the main steps followed in the analysis, the level of effort, the difficulties and the main findings. The second section looks at the analyses from the Regulator's standpoint. after review of most of the assessments has taken place. Finally, the last section of the paper contains a general discussion about the whole process and the usefulness of the results in the general context of probabilistic safety assessments (PSA).

SEISMIC MARGIN ANALYSIS OF SPANISH PLANTS

General comments

There are two main approaches for assessing the seismic risk of a nuclear unit: the seismic probabilistic risk analysis (seismic PRA) and the seismic margin assessment (SMA) In both cases the main purpose is the same: to understand the most likely severe accident sequences induced by earthquakes and to identify the dominant seismic risk contributors or seismic weaknesses. However, the methodology and the results produced by these two general approaches are different.

The seismic PRA, is an extension of the classical event tree/fault tree methodology used for probabilistic safety analysis with "internal" initiators. In the seismic PRA, instead of dealing just with random equipment failures, the earthquake appears as a new cause for failure; and the frequency of failure of a particular component is computed from the seismic hazard of the site (probability of the earthquake) and the fragility of the component (conditional probability of failure). The basic results are the same as in classical PRA methodology: the accident sequences leading. to core damage and the frequency of each of those. The first seismic PRAs were published by the end of the seventies (Ref. 5) arid required significant resources. This motivated a research effort for a less expensive methodology. This efort took advantage both from the number of seismic PRAs already carried out and from the Seismic Safety Margins Research Program (Ref: 6. 7). As a result, the so-called seismic margin methods (SMA) were developed (Ref, 8. 9, 10. 11). In Spain. the PRA option was not followed and all plants selected an SMA approach for the seismic risk analysis.

The main difference between seismic margin methods and seismic PRAs is that the former do not use a seismic hazard study of the site, which is usually highly controversial (Ref. 4). Instead of looking for a core damage frequency, seismic margin studies look for the level of earthquake below which core damage is .very unlikely (figure 1). This level of earthquake is called "high-confidence, low-probability of failure" (HCLPF) capacity of the plant. If a seismic hazard study of the plant is available, an estimate of the seismic contribution to core damage frequency can be computed straightforwardly from the HCLPF capacity.

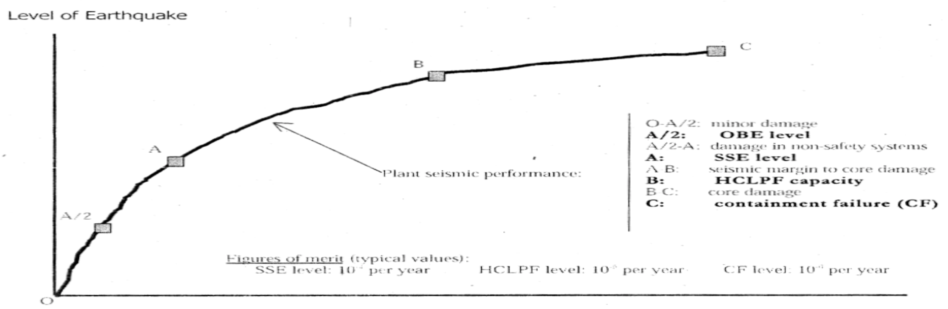

Figure 1. The concept of seismic margin

Another difference between seismic PRAs and seismic margin methods is that, in the latter. the systems analysis part is much simpler. In a seismic margin study only some "basic safety functions" are taken into account. since the experience of seismic PRAs has shown that when these basic safety functions are preserved, core damage and radioactive release is very unlikely.

Both methodologies, seismic PRA and seismic margin methods, benefited from the American industry enormous effort to resolve the USI A-46 (Unresolved Safety Issue A-46: "Verification of Seismic Adequacy of Equipment in Operating Plants"). This issue affected a significant portion of American units, those with construction permits docketed before about 1972, in which the equipment had not been seismically qualified according to present standards. Industry formed the SQUG (Seismic Qualification Utility Group) and financed the development of a practical methodology to verify the seismic adequacy of installed equipment. During the eighties. an alternative qualification method and the corresponding acceptance criteria were conceived and prepared for practical use. The method, gathered in the GIP (Generic Implementation Procedure, Ref. 12), is largely based in the experience on the effects of actual earthquakes in real equipment (Ref. 13) and provides the analyst with a very efficient way of assessing seismic capacity. The rules of the GIP, and the use of seismic experience in general, simplified and added efficiency to seismic PRAs and seismic margin studies. At the same time, the recourse to seismic experience showed the importance of plant walkdowns by seismic and system engineers. Plant walkdowns are now the key component both in seismic PRAs and seismic margin studies.

As mentioned earlier. research to develop a seismic margin review methodology began under NRC support at the Lawrence Livermore National Laboratory in 1984. An expert panel vas formed to develop the methodology and this panel issued its reports In the following years (Ref. 8. 9). A parallel effort was undertaken by the Electric Power Research Institute (EPR1), and the EPRI methodology that resulted (Ref. 11) is slightly different from the NRC approach. The methodology corning, from LPRI is very well documented in Ref 11 and that makes its application easier. Basically, all Spanish plants chose the EPRI seismic margin methodology to carry out the seismic risk assessment.

Seismic margin assessment (SMA)

A general definition of seismic margin is expressed in terms of the earthquake motion level that compromises plant safety. specifically leading to melting of the reactor core. Operationally, the margin is expressed in terms of the difference between the earthquake notion level that compromises plant safety and some smaller motion level, such as the plant's earthquake design basis (figure l). The measure of seismic capacity adopted in seismic margin reviews is the so-called "high confidence, low probability of failure" capacity, or HCLPF capacity, usually 'given in units of peak ground acceleration. This is a conservative representation of capacity and, in simple terms, corresponds to the earthquake level at which it is extremely unlikely that failure of the component will occur. From the mathematical perspective of a probability distribution of capacity developed in seismic PRA calculations, the HCLPF capacity values are approximately equal to a 95 confidence (probability) of not exceeding about a 5% probability of failure. Using the HCLPF concept, the search for the seismic margin shifts to determining the plant-level HCLPF capacity and comparing it with the design basis earthquake. The flowchart of the review. as implemented by Spanish plants. can be seen in figure 2.

As mentioned earlier, for the Spanish plants, the seismic margin earthquake (SME), also called review level earthquake (RLE), was assigned to each plant by the CSN: For all plants, the RLF was a 0. 3 ZPGA earthquake, whose frequency content was defined by the median (50% exceedance probability) response spectrum of NUREG/CR-0098 (Ref. 14). It should be mentioned that the RLE is just a screening tool for the review, in the sense that the seismic capacities of individual components are compared with the RLF and deemed not controlling plant margin if they can withstand the RLL. With the help of' screening tables

(Ref. 8, 11), it is easier to compare the capacities of the components with a given earthquake motion than to compute the capacity of every component of the plant.

Once the RLE is defined, the next step is developing structural response and floor spectra. In Spanish plants this task has invariably been done by scaling design (SSE) response and floor spectra. This has been an acceptable procedure since all sites can be considered "rock" sites. For each building, a scale factor to convert from SSE to RLE has been determined. In computing this factor two sources of conservatism have been eliminated: the structural damping values have been adjusted to higher, more realistic figures and spatial horizontal incoherence of seismic waves has been considered for frequencies above 5 Hz (Ref 11).

In the other branch of the flowchart (figure 2), there is a system analysis. 1n the EPRI methodology, neither event trees nor fault trees are used. The methodology works with the concept of "success path". A success path is a way of bringing the plant to a .stable condition (either hot or cold shutdown) and maintaining that condition for at least 72 hours. .According to the methodology, the four principal safety functions that are required to achieve and maintain a safe shutdown condition are: reactivity control, reactor coolant system pressure control, reactor coolant system inventory control and decay heat removal. Additionally, NUREG-1407 (Ref. 2) requires consideration of containment isolation.

The various means of accomplishing these safety functions depend on the plant's design. The analyst must determine two success paths, a preferred path and an alternative path, as independent as possible and covering the above safety functions. For both paths, unrecoverable loss of off-site power (LOOP) has to be assumed and at least one of the paths has to be able to cope with a small LOCA (l" diameter equivalent break). The main result of the system analysis phase of the margin assessment is a list of components subject to evaluation: The components in the list are those necessary to implement the success paths. The list include equipment, distribution systems, such as piping and cable trays, and structures. The list of equipment is usually known as the "safe shutdown equipment list" (SSEL).

Most of the Spanish plants have used the previous work in level 1 PRA to determine the SSEL. The components in the list are obtained from the basic events in the fault trees corresponding to the event trees with LOOP and small LOCA as initiating events. The list is completed with some passive components, such as structures, and with the equipment needed for containment isolation. In a typical Spanish plant, the SSEL has had between 300 and 400 items.

After the SSEL has been compiled and the RL.E seismic demand has been determined in the form of floor spectra, the flow of the study converges into the preparation of the walkdown. In Spanish plants the purpose of this phase has been to prepare plant walkdown by carrying out a preliminary assessment of component and anchorage capacities. The goal is to support judgment of the walkdown team by supplying them with information about the configuration of the components, the anchorage, the seismic qualification procedures and a assessment of margins above the design basis earthquake. Gathering this inl6rmation has proved to be lengthy and sometimes difficult, especially in the oldest plants.

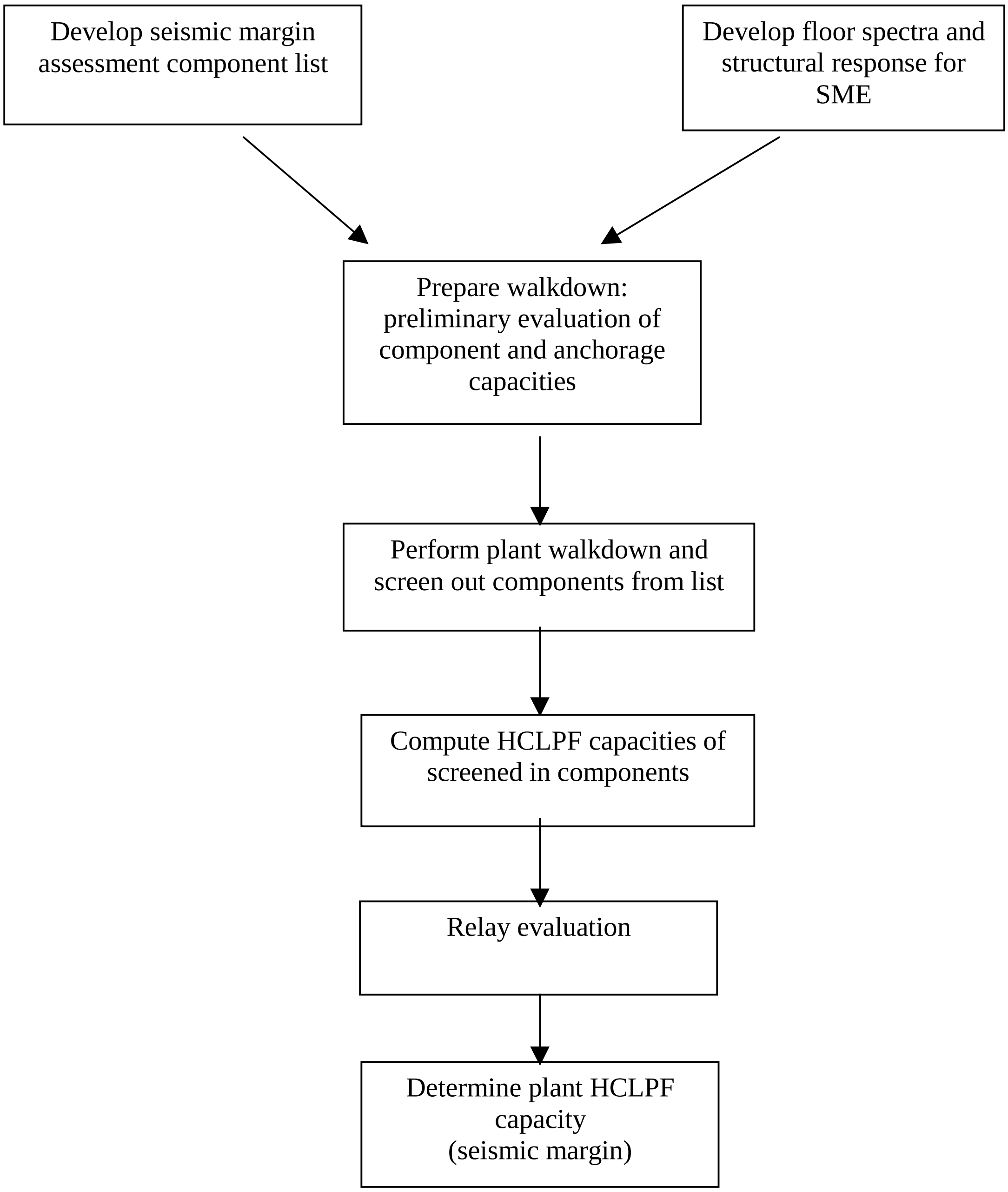

Figure 2. Flowchart of a typical seismic margin assessment for a Spanish plant

During this phase, information about the general seismic design of the plant has been reviewed. Also, information about each of the components in the evaluation list has been processed. Typically, for each component. configuration and anchorage drawings and seismic qualification reports have been gathered. Then, from this information and from the RLL. floor spectra developed in the previous phase. a preliminary capacity and anchorage assessment has been carried out and documented. Only very simple bounding computations have been performed. However, these simple computations, sometimes supported by seismic experience rules (Ref. 12, 13), have been usually enough to conclude that many items in the list have capacities above the RLE.

The plant walkdown has been taken as the key phase of the whole methodology. During this phase a walkdown team that includes structural and system engineers has inspected all the components in the evaluation list that were reasonable accessible and located in a low to moderate radiation environment. The team judges whether or not the HCLPF capacity is higher than the RLE. The basis for this judgement is in the screening tables included in Ref. 8 and 11, and in the preliminary assessment work carried out before the walkdown. Components whose HCLPF capacity is judged to be higher than the RL.E are considered not to control HCLPF capacity of the plant and, hence, screened out front the evaluation. For components not screened out, failure modes are clearly defined for the HCLPF capacity evaluation in the next phase of the study. Plant walkdown has always been documented according to reference 11. Documentation includes an evaluation sheet for each of the inspected items ("seismic evaluation work sheet" or SEWS) and a list that summarizes the SEWS ("seismic verification data sheets" or SVDS). In a typical Spanish plant the plant walkdown has taken between two and three weeks of site work, with a team of about four people. Usually, an American senior seismic consultant (Dr. Robert Kennedy, from RPK, or Dr. Mayasandra Ravindra, from [EQE) has participated in some part of the walkdown (one week).

After the walkdown, usually only a small number of components were screened in and; therefore, remained in the evaluation list. These components were assumed to control the seismic margin of the plant. The purpose of the neat phase was to compute HCLPF capacity of these components according to the failure modes specified by the walkdown team in the SEWS. In Spanish plants, the number of components for which HCLPF calculations were needed varied from about 15 to =10. Once the I ICLPF capacity of the components screened in during the walkdown were computed, it was postulated that the HCLPF capacity of the plant as a whole is given by the component with smaller capacity.

In addition to the HCLPF capacity of the plant, the main result of the SMA is the list of components with HCLPF capacity smaller than the RLE. These are the seismic weak links of the plant and they are the focus of any modification program with the goal of increasing the seismic safety of the installation. Decisions on whether to introduce changes or not have been taken on a cost/benefit basis.

Plant	Reactor type	Power (MWe)	Operation since	SSE	HCLPF capacity	Seismic weak links
José Cabrera	Westinghouse PWR	160	1968	0.07 g	0.16 g	Vertical tanks
Sta Maria de Garona	GE-BWR	466	1971	0.10 g	0.17 g	Vertical tank
Almaraz I	Westinghouse PWR	974	1981	0.10 g	0.20 g	Vertical links
Ascó I	Westinghouse PWR	1028	1982	0.13 g	0.16 g	Inverters, relays
Almaraz II	Westinghouse PWR	983	1983	0.10 g	0.20 g(*)	Vertical tanks
Cofrentes	GE-BWR	994	1984	0.17 g	0.28 g	Relays
Ascó II	Westinghouse PWR	1024	1985	0.13 g	0.16 g	Inverters, relays
Vandellós II	Westinghouse PWR	1004	1988	0.20 g	>0.30 g	Vertical tanks
Trillo	Siemens KWU-PWR	1066	1988	0.12 g	not available	-

(*) Preliminary result

Table 2. Seismic margin assessment of Spanish power plants

Results

In Spanish plants the overall level of effort for an SMA has been in the order of 5 man-year. The results obtained for all Spanish plants are given in table 2. The first thing that should be said is that in the vast majority of plants only a small number of components were found to have a HCLPF capacity below 0.3g. This is so even for plants with a relatively low design basis earthquake (SSE). In the SMAs performed for Spanish plants, the average number of components with a HCLPF capacity below 0.3g is about five and they usually include vertical atmospheric tanks, electrical equipment, such as relays, and anchorage details.

On the other hand, the walkdown usually has found many housekeeping issues related with seismic spatial interactions and a considerable amount of small things to be fixed. Such as missing bolts, untied electrical cabinets, etc. These small repairs have been usually carried out by the utility shortly after the walkdown.

REVIEW BY THE REGULATORY STAFF

The Spanish regulator (CSN) is responsible for licensing all activities related with the seismic risk analysis of Spanish plants. The utilities have documented the work carried out in accordance to the requirements of Ref. 2, and in each case a final report has been submitted to the CSN for review. The procedure used by the CSN in the licensing activities includes three major items:

- Evaluation of the documentation submitted by tile utilities. In this step the methodology, general criteria, development of the component list, screening process, walkdown results, engineering judgments, Fragility analyses, peer review, component and plant HCLPFs, uncertainties, etc.. as have been described on the documentation, are evaluated.

- Agreement on methodology. The CSN evaluation team together with the utility SMA team interchange points of view and contributions about the major issues observed during the process. The objective is to reach an agreement on the assumptions, uncertainties, sensitivity and engineering judgments, etc.. matching the different opinion.
- Joint walkdown. The purpose is to examine the general seismic state of the 'plant, the improvements introduced as a consequence of the seismic assessment activities, etc. This is done together with the utility team.

From a regulator's point of view, the main results of the seismic margin analyses have been the true knowledge of the actual seismic capability of each Spanish NPP and the identification of the weakest links from a seismic standpoint. In general, according to the Spanish seismic hazard current state of knowledge and before the final outcome from CSN review, all Spanish plants would have enough capacity for safe operation during and after an earthquake far beyond the design basis (SSE), as shown in table 2 (HCLPF capacity). This includes the older plants, which previously to the IPEEE program had gone through a process of seismic revaluation of systems, structures and equipment (SEP, USI A-46). In all plants the seismic weak links have been identified, their HCLPFs have been obtained and, when feasible, modifications to improve overall seismic capacity of the plant have been proposed. At the moment, these improvements are being. implemented.

Nevertheless, when the CSN has carried out the plant walkdown, usually some years after the official walkdown, housekeeping problems similar to those found during the official walkdown, when not the same, have been found. This is the case of unanchored equipment near cabinets with essential relays, equipment on wheels near safety components, new cabinets placed close to safety cabinets without being tied to them, gas bottles without seismic ties, cabinet doors left opened. etc. This is a the result of the absence of some kind of seismic training for maintenance and operation staff. This could be also a consequence of the lack of involvement of the utility staff in the seismic revaluation process.

DISCUSSION

Spain is country where a general awareness of the seismic risk does not exist. This is true trot only in the nuclear industry, but also in many other fields. The last earthquake with casualties took place in 1884. Hence, there is no living generation with a direct experience in this kind of natural phenomenon. As a consequence, any attempt to improve seismic safety has to light with the feeling that "earthquakes do not happen in Spain" and "there are better places to put the money in".

The lack of seismic awareness is in the root of all the difficulties in developing the seismic portion of the Integrated Program for Probabilistic Safety Assessment' in Spain. As in the case of NRC, the Spanish Regulator's objective has been to find the vulnerabilities to earthquakes above the SSE level, and to prove that even if an earthquake beyond SSE occurs, the core damage frequency or the radioactive material release frequency are below $l0^{-5}$ and 10^{-6} per year, respectively. On the other hand, the utilities have always considered that the level of seismic safety in the plants is more than enough for a county with a low to moderate seismicity and that the effort required to find the weak links beyond the SSE level is not justified on a cost-benefit basis. Due to the absence of strong shocks during the last century, the results of the seismic margin assessments performed in Spain seem to support the thesis of the utilities: the conservatism of design has produced, in most cases, a wide margin of safety above SSE.

In any case, in the view of the authors, the work for the Spanish plants has showed that a seismic margin assessment is a cost effective method for a periodic revaluation of the seismic safety of a given plant. It allows both to find out its actual state in terms of seismic vulnerability and to prevent degradation of its seismic safety. In this line, seismic walkdowns have detected many "minor issues", immediately corrected by maintenance staff, that otherwise would have contributed to a bad seismic performance. The experience shows that these minor issues tend to reappear periodically. This is a consequence of not having, maintenance people permanently looking at the plant with seismic eyes.

On the other hand, from a regulator's point of view, the SMA is tool that allows a straightforward evaluation of the contribution to the overall risk that derives from a revised seismic hazard at the site. The revision of seismic hazard will be the consequence of the continuous research or advances in the seismological knowledge, combined with the occurrence of earthquakes during the life of the plants. The capacity of the weak links of the plant, together with the seismic hazard at the site, can be used to determine the core damage frequency derived from earthquakes and its contribution to the total risk posed by the plant. In this sense, it seems advisable to keep the methodology alive. updating the results with plant changes.

CONCLUSIONS

The conclusions could be summarized as follows:

- A seismic margin assessment is a cost effective method to determine the plant state regarding seismic safety and to identify the most effective countermeasures on a cost-benefit basis.
- A periodic update of the assessment will help prevent plant safety degradation due to changes in the plant and replacement of components.
- From a regulator's point of view, the seismic margin assessment is tool that allows a straightforward evaluation of the overall risk derived from the seismic hazard at the site.
- In Spain the utilities are generally under the impression that this kind of studies do not have an appropriate cost-benefit ratio, that is, the findings do not justify the investment This impression could be a consequence of the general lack of awareness of the seismic risk in the Spanish society.

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