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SEISMIC SAFETY OF NUCLEAR POWER PLANTS

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Abstract

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

1. INTRODUCTION

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

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

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

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

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

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

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

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

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

2. REVIEW SERVICES

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

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

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

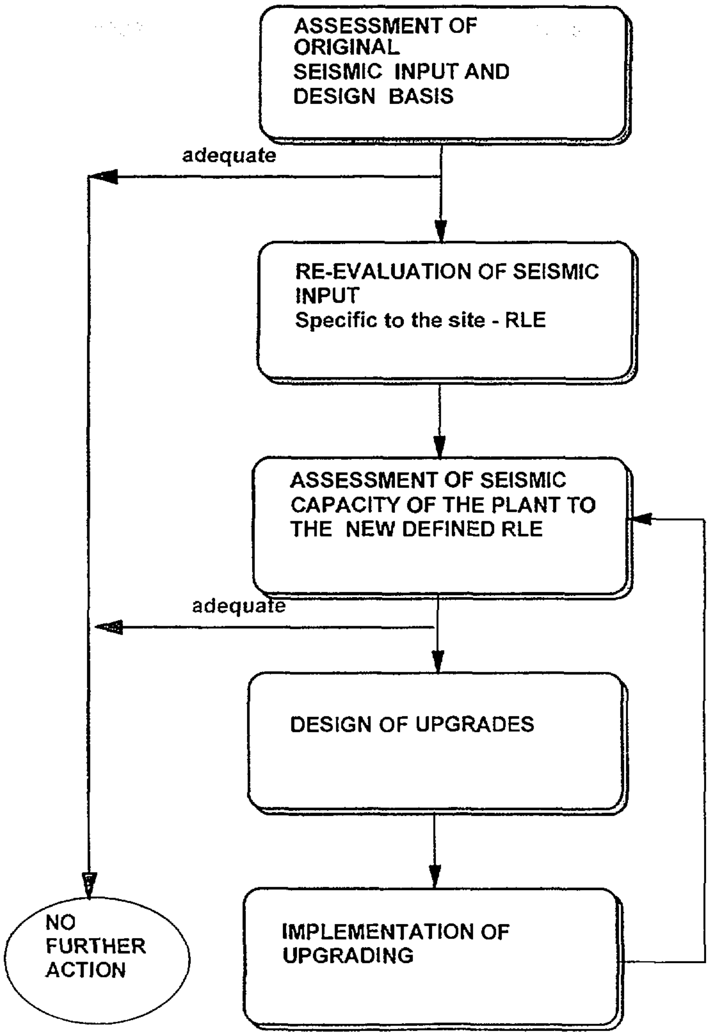

FIG. 1. Flow diagram for seismic re-evaluation and upgrading of existing nuclear power plants.

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

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

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

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

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

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

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

Legend:
SDB: Seismic Design Basis
NED: No Explicit Design
Neg.: Inadequate seismic capacity for the reassessed SDB (RLE)
?: A question mark indicates an ongoing activity with a preliminary indication of the reassessed SD13 (RLE)
No: The activity has not started yet
*: Incomplete.

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

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

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

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

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

3. CRITERIA FOR RE-EVALUATION OF SEISMIC CAPACITY

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

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

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

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

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

(b) safe shutdown means hot or cold shutdown;

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

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

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

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

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

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

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

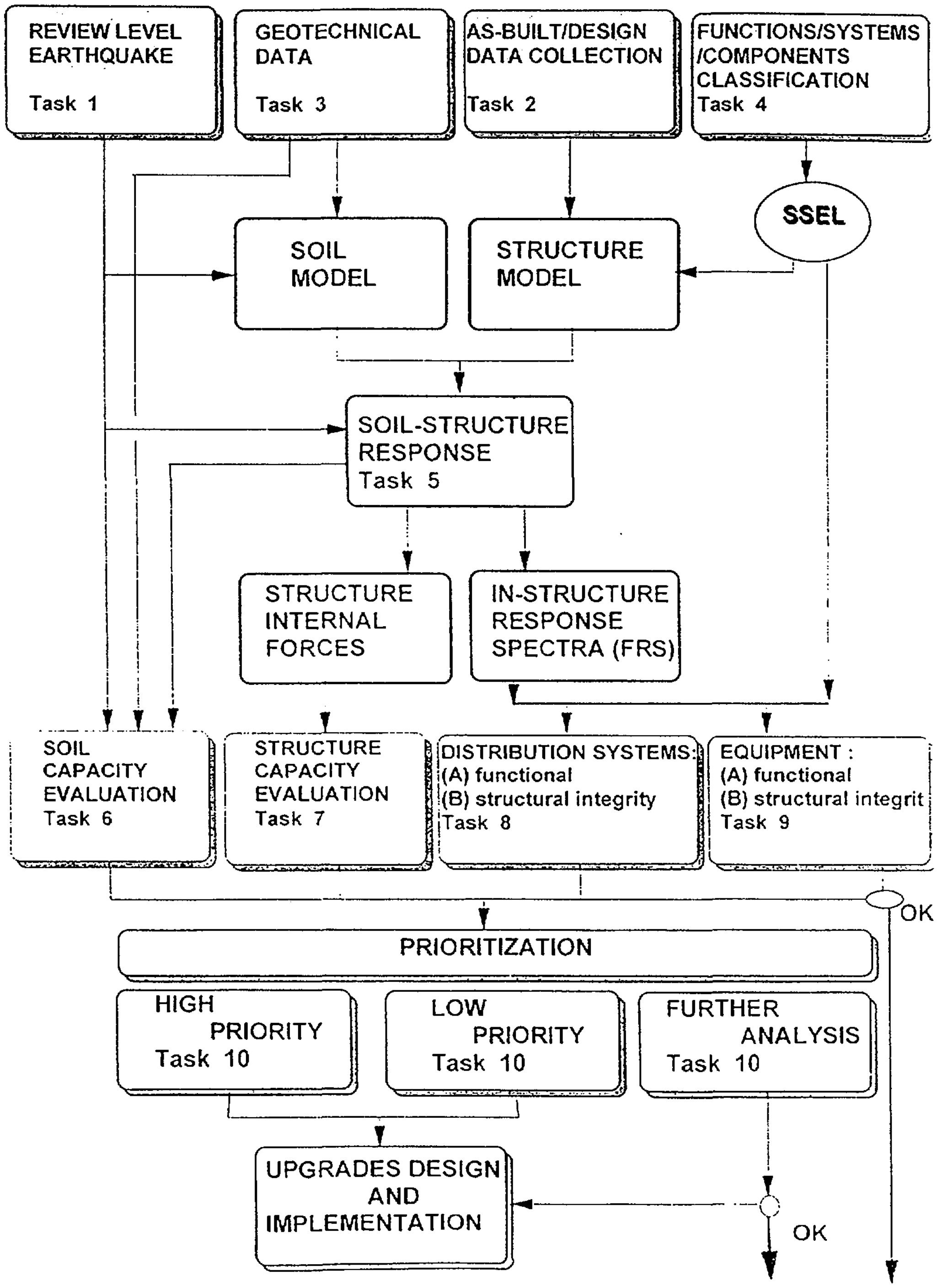

FIG. 2. Detailed flow diagram for the assessment and improvement of seismic safety.

3.2. Plant walkdown

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

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

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

(1) Disposition 1: a fix is required;

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

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

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

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

3.3. Evaluation of seismic margin capacity

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

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

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

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

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

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

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

TABLE III. PARTITION OF TASKS FOR PARTICIPATING INSTITUTIONS

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

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

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

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

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

Notes to Table V:

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

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

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

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

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

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

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

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

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

Step 1: calculate elastic seismic demand in members and connections by elastic seismic response analysis, using the elastic response spectrum;

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

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

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

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

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

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

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

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

4. CO-ORDINATED RESEARCH PROGRAMMES

4.1. Background

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

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

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

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

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

4.2. Prototype plants

Paks NPP

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

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

Kozloduy NPP Unit 5/6

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

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

4.3. Participation and tasks

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

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

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

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

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

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

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

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

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

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

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

Task 7a. Reactor building (task continuing)

Task 7b. Stack (task continuing)

Task 7c. Worm tank (task continuing)

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

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

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

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

Task 11. Previous component data (task continuing)

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

Volume 1. Data related to sites and plants (Pales and Kozloduy NPPs)

Volume 2. Generic material: codes, standards, criteria

Volumes 3A, 313, 3C, 3D, 3E. Kozloduy NPP, Units 5/6: Analysis/testing

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

Volumes 4A, 5B. Experience data

4.4. Full scale dynamic test of Paks and Kozloduy NPPs

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

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

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

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

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

5. CONCLUDING REMARKS

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

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

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

ACKNOWLEDGEMENT

The authors acknowledge that parts of this article have been taken from technical material contributed by various consultants in the course of the past seven years either in relation to IAEA Engineering Safety Review Services for external events or the Coordinated Research Programme on "Benchmark study for the seismic analysis and testing of WWER type NPPs".

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

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[2] NUREG/CR-0098, "Development of Criteria for Seismic Review of Selected Nuclear Power Plants", N.M. Newmark and W.J. Hall, NRC, May 1978.
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ANNEX 1

ENGINEERING SAFETY ADVISORY SERVICES
Related to Site and External Hazards

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

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

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

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

S: Review, of site investigations for all disciplines involved.

S-F: Follow-up mission of previous reviews of site investigations.

SI: Review of investigations for determining the seismic input parameters, specific to the site

SI-F: Follow-up mission of previous reviews of seismic input definition.

SC: Review of seismic capacity and necessary upgrading of systems. structures and components (SSC) of the plant.

SC-F: Follow-up mission of previous reviews of seismic capacity and upgrades of SSC.

W: Workshop.

WP: Review of work plans and technical procedures for the site and seismic safety assessment.

B: Activities related to benchmark project for seismic safety of WWER type NPPs.

ENGINEERING SAFETY ADVISORY SERVICES Related to Site and External Hazards Year 1995					
NO.	**TYPE**	**COUNTRY**	**NPP/LOCATION**	**DATE**	**PLANT TYPE**
88.	SC-F	Pakistan	Chashma	Janu 95	PWR 300
89.	SI-F	Hun ary	Paks	Janua 95	WWER-440/213
90.	S-F	Indonesia	Muria Peninsula	March 95	not defined vet
91.	SC-F	Bulgaria	Kozlodu -5	March 95	WWER-1000
92.	SC-F	Slovakia	Mochovce	April 95	WWER-440/213
93.	SI-F	Hungary	Paks	April 95	WWER-440/213
94.	SI-F	Armenia	Medzamor	April 95	WWER-440/230
95.	S	Czech Rep.	(not defined vet	Ma 95	Se pent Fuel Storage
96.	S	Thailand	not defined et	May 95	not defined yet)
97.	SI-F	Armenia	Medzamor	May 95	WWER-440/230
98.	SI/SC-F	Kazakhstan	Alma Ata	May 95	WWER-10 Res. Reactor
99.	SI/SC	Uzbekistan	Tashkent	May 95	WWER-10 Res. Reactor
100.	SI-F	Hungary	Paks	June 95	i WWER-440/213
101.	B-W	Russia	(not applicable)	June 95	W'WER type reactor
102.	S-F	Indonesia	Muria Peninsula	July 95	(not defined yet)
103.	S	Thailand	site selection process)	July 95	(not defined yet)
104.	SI-F	Bulgaria	Belene	July 95	WWER-1000
105.	S-F	Morocco	Sidi Boulbra	September 95	WWER-1000
106.	SC-F	Pakistan	Chashma	September 95	PWR 300
107.	S-F	Indonesia	Muria Peninsula	November 95	(not defined yet)
108.	S	Thailand	(site selection process)	November 95	not defined vet
109.	SI	Iran	Bushehr	December 95	PWR 13-WWER-1000
110.	SI-F	Hun a	Paks	November 95	W WER-440/213
111.	W	Korea	Regional	December 95	Generic
112.	SI-F	Czech Rep.	Skalka	December 95	Sent Fuel Storage

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SI: Review of investigations for determining the seismic input parameters, specific to the site

SI-F: Follow-up mission of previous reviews of seismic input definition.

SC: Review of seismic capacity and necessary upgrading of systems, structures and components (SSC) of the plant.

SC-F: Follow-up mission of previous reviews of seismic capacity and upgrades of SSC.

W: Workshop.

WP: Review of work plans and technical procedures for the site and seismic safety assessment.

B: Activities related to benchmark project for seismic safety of W WER type NPPs.

ENGINEERING SAFETY ADVISORY SERVICES Related to Site and External Hazards Year 1996					
NO.	**TYPE**	**COUNTRY**	**NPP/LOCATION**	**DATE**	**PLANT TYPE**
113.	WP	Armenia	Medzamor	January 96	WWER-440/230
114.	SI-F	Hungary	Paks	January 96	WWER-440/213
115.	WP	Slovakia	Bohunice	January 96	WWER-440/213
116.	SI-F	Slovenia	Krsko	February 96	PWR 600
117.	SC	Slovenia	Krsko	February 96	PWR-600
118.	W	Armenia	Medzamor	March 96	WWER-440/230
1 19.	S-F	Indonesia	Muria Peninsula	April 96	(not defined yet)
120.	B	Bulgaria	Kozloduy	June 96	WWER-1000
121.	B-W	Italy	(not applicable)	June 96	W WER type reactors
122.	SC-F	Pakistan	Chashma	June 96	PWR 300
123.	SC-F	Armenia	Medzamor	July 96	WWER-440/230
124.	SC	Armenia	Medzamor	September 96	WWER-440/230
125.	W	Korea	Generic	September 96	-
126.	SC-F	Armenia	Medzamor	November 96	W WER-440/230
127.	S-F	Indonesia	Muria Peninsula	December 96	(not defined yet)

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

ENGINEERING SAFETY ADVISORY SERVICES RELATED TO SITE AND EXTERNAL HAZARDS (ESRS)

YEAR	TOTAL NUMBER OF MISSIONS
1989	5
1990	8
1991	14
1992	14
1993	23I
1994	23
1995	25
1996	15
TOTAL	127

NUMBER OF EXTERNAL EXPERTS PER YEAR: 76
(AVERAGE OF LAST TWO YEARS)

IAEA RESOURCES:
1 STAFF MEMBER
1 STAFF MEMBER AS TA

ANNEX 2

DRAFT WORKPLAN
1996/97

Task 1. **Safe shutdown systems identification/classification**
Paks NPP (WWER-440/213) - **Task completed**
Kozloduy NPP (WWER-1000) - **Task completed**
(Co-ordinated by WESE)

Task 2. **Design regulations, acceptance criteria, loading combinations**
AEP will provide FRS for Kozloduy NPP. Otherwise **task completed.**

Task 3. **Seismic input, soil conditions**
Task completed for Kozloduy NPP.
Final soil parameters and seismic input will be sent by Paks NPP **(June 1996)**

Task 4. **Standards, criteria - comparative study**
First phase completed. For the second phase comparison of re-evaluation criteria will be made with original Soviet rules at the time of design. **(October 1997)**

SA (CR) - coordinator
MD (CR)
CKTI (RF)

Task 5. **Walkdown of reference plants (Paks and Kozloduy Unit 5)**
Task completed for both plants.

Task 6a. **Dynamic analysis of Kozloduy NPP Unit 5 RB for seismic input**

AEP (RF) - Siemens - Coordinator
EQE (BG)
MD(CR)
CL(BG)

Task completed.

Task 6b. **Dynamic analysis of Paks NPP RB for seismic input**

AEP (RF) - Siemens (G) - coordinator MD (CR)

Task completed.

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

Task 7a **- Reactor building**

Participants: Siemens (G)
EQE (BG)
CL (BG)
MD (CR)
IVO (F)

Referee: Ismes (I)

Input: Distributed by IAEA.

Soil Data: Revised soil data to be distributed by Paks NPP (**June 1996**)

Output: Indicated points on the basemat (4-6, 14-16, 21, 33), at elevation +18.9 (35-37) and on the steel structure (46-47).

Response parameter: acceleration time histories

Format: on diskette or b;, e-mail to Mr. Zola (Ismes) plus a hard copy

Transmittal of "response" by participants (September 1996)

Comparative report by Ismes (December 1996)

Task 7b. **Stack**

Participants: SAGE (B)
IZIIS (M)

IZIIS (M) will prepare final report. **(October 1996)**

Task 7c. **Worm tank**

Participants:
AES (US) (in cooperation with Japanese institutes)
PNPP (H)
SA (CR)

AES (US) will evaluate results of experiments conducted so far. **(October 1997)**
PNPP (H) will process blast results and compare with shaking table test results. **(October 1997)**
SA (CR) will study seismic behavior of the tank for sliding. **(October 1997)**

Task 8a. **Full scale blast testing of Paks NPP**

Participants: PNPP (H) and Ismes (I)

Task completed.

Task 8b. Full scale blast testing of Kozloduy NPP Unit 5

Participants: KNPP (BG), CL (BG), Ismes (1) **(July 1996)**

Task 9. Shaking table experiment for selected components
Participants: IZIIS (M) - coordinator, KNPP (BG), PNPP (H)

Five different types of relays will be tested each from Kozloduy and Paks NPPs.

Delivery of relays: **(September 1996)**
Testing: **(December 1996)**

Task 10. On site testing of equipment at Paks and Kozloduy NPPs

Participants: VNIIAM (RF) - coordinator, PNPP (H), KNPP (BG)

Task continuing.

Task 11. Previous component test data

Participants: IZIIS (M), EQE (BG), KNPP (BG), PNPP (H), AEP (RF), VNIIAM (R-F), CKTI (RF), SA®

SA(R) will compile a list using the information in the Working Material provided by the other participants as well as Eurotest. (December 1996)

Task 12. Experience data from Vrancea and Armenia earthquakes

Participants: AEP (RF), SA (R), EQE (US)

Task continuing for Vrancea data. **(December 1996)**

Task 13. Experience data from US earthquakes

Participants: EQE (US), WESE (B), SA (US)

Task continuing. EQE (US) and SA (US) will meet and discuss in two weeks.

Task 14. Special Topic 1 - Assessment of containment dome prestressing for KNPP

Participants: KNPP (BG), SP (CH), BRI (BG)

New tendons will be designed using Swiss technology, material for 10 tendons for the cylindrical part of the RB will be delivered, a new monitoring system will be evaluated and implemented. **(October 1997)**

Task 15. Special Topic 2 - Assessment of containment dome/cylindrical part for different loading combinations

Participants: KNPP (BG), SP (CH), BRI (BG), EQE (BG)

Task continuing. **(June 1997)**

Task 16. Special Topic 3 - Stress analysis of safety related piping for KNPP

Participants: SP (CH) - co-ordinator, KNPP (BG), Woelfel (G), CKTI (RF)

Task for Woelfel completed. SP (CH) will perform the following: seismic capacity evaluation of remaining piping systems, recommendations for upgrade measures, redesign of support structures where upgrades are necessary, and implement new support structures. **(October 1997)**

Task 17. Special Topic 4 - Dynamic analysis of selected structures of KNPP

Participants: SP (CH) - coordinator, KNPP (BG), BRI (BG), EQE (BG)

Diesel generator building analysis finished by BRI, interaction with underground reservoirs is in progress. (October 1997)

EQE (BG) submitted report on stack to KNPP who will transmit to IAEA.

Other stack analysis finalized by SP (CH).

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

Participants: PNPP (H), CKTI (RF), SA (CR), WESE (B)

Task continuing. **(December 1996)**

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

Participants: EQE (US), Siemens (G)

Task continuing. **(October 1997)**

Task 20. Analysis of buried pipelines for PNPP

Participants: SAGE (B), Siemens (G), SA (CR)

Task continuing. **(October 1997)**

Task 21. Comparison of beam vs 3D models for KNPP and PNPP structures

Participants: MD (CR), EQE (BG)

Task continuing. **(October 1997)**

Task 22. Experience data base (WWER SQUG) initiation

Participants: PNPP (H), KNPP (BG), EQE (US), SA (US), SA (R), SA (CR)

A format will be prepared by SA ® and SA (CR). KNPP and PNPP will check feasibility of providing a sample for database. (December 1996)