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UPGRADING OF SEISMIC DESIGN AND INVESTIGATION ON AGEING ISSUE OF NUCLEAR POWER PLANTS IN JAPAN

H. SHIBATA
Association for the Development of Earthquake and Prediction

Y. KITADA
Seismic Engineering Center,
Nuclear Power Engineering Corporation

Japan

Abstract

In Japan, seismic design methodology of nuclear power plant (NPP) has been established [1]-[3]. And yet efforts have been continued to date to upgrade the methodology, because of conservative nature given to the methodology in regard to unknown phenomena and technically-limited modeling involved in design analyses. The conservative nature tends to produce excessive safety margins, and inevitably send NPP construction cost up. Moreover, excessive seismic design can increase the burden on normal plant operation, though not necessarily contributing to overall plant safety. Therefore, seismic engineering has put to many tests and simulation analyses in hopes to rationalize seismic design and enhance reliability of seismic safety of NPPs. In this paper, we firstly describe some studies on structural seismic design of NPP underway as part of Japan's effort to upgrade existing seismic design methodology. Secondly we introduce a summary of an investigation performed in Japan to investigate the effect of aging of NPP structures and equipment on the seismic safety of an NPP.
Most studies described here are carried out under the sponsorship of MITI (the Ministry of International Trade and Industry Japan), though, similar studies with the same motive are also carrying out by nuclear industries such as utilities, NPP equipment and system manufacturers and building constructors.
This paper consists of three sections, each introducing studies relating to NPP structural seismic design, upgrading of the methodology of structural design analyses and investigation to establish an evaluation methodology of aging effect on seismic safety of an NPP..

1. STUDIES ON STRUCTURAL SEISMIC DESIGN

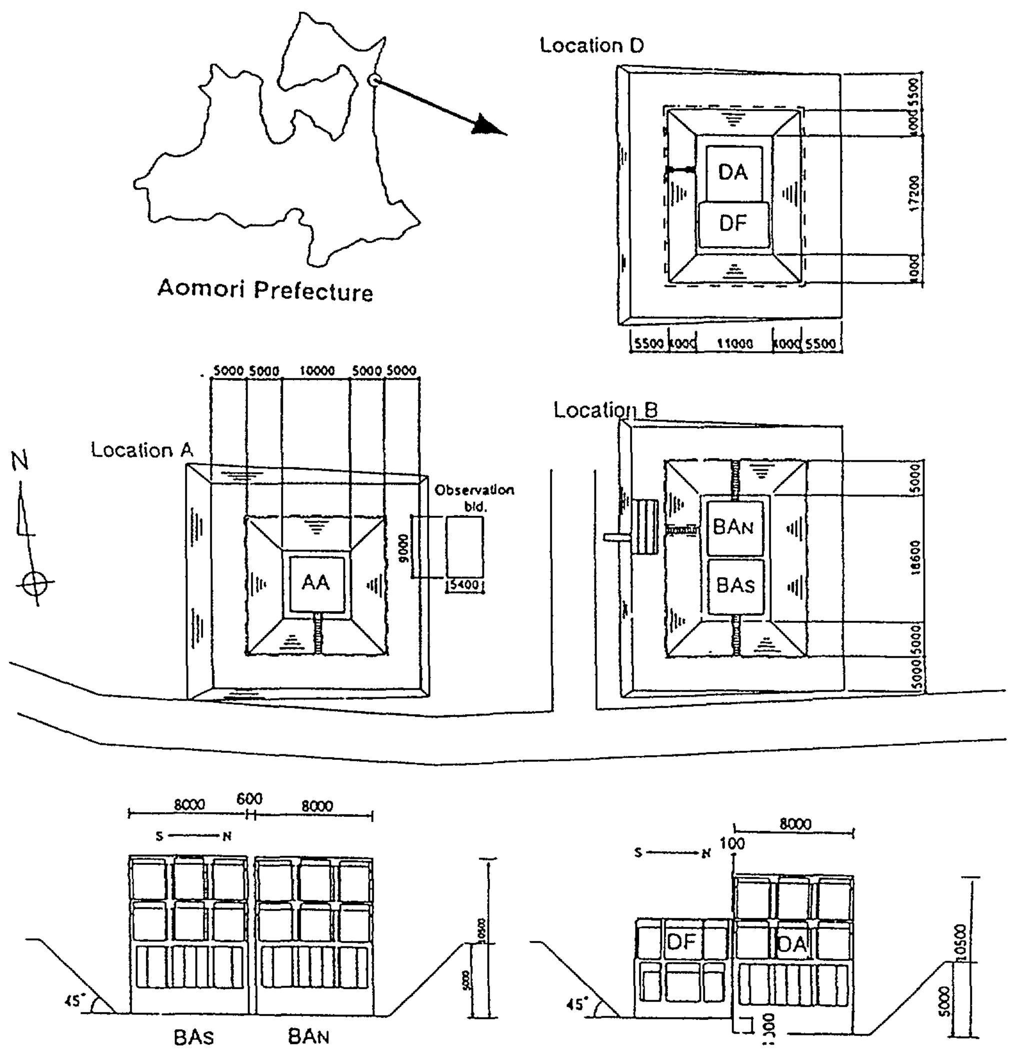
In this section, following four studies are introduced as typical examples of ongoing studies on upgrading of NPP structures ;

(1) Model Test of Dynamic Cross Interaction Effects of Adjacent Structures,
(2) Model Test of Multi-axes Loading of RC (reinforced concrete) Shear Walls,
(3) Seismic Proving Test of Concrete Containment Vessels,
(4) Application Study on Seismic Base Isolation System to An NPP Building.

1.1. MODEL TEST OF DYNAMIC CROSS INTERACTION EFFECTS OF ADJACENT STRUCTURES

The objective of this test is to clarify the effects of structures built adjacent to an NPP reactor building, i.e., a turbine building etc., on the dynamic characteristics of the reactor building because, in

building, i.e., a turbine building etc., on the dynamic characteristics of the reactor building because, in the current seismic design analysis, the reactor building is modeled as single independent structure. Actually, many massive and heavy buildings are constructed close to a reactor building i.e., another reactor building, turbine building etc. This effect is categorized as the structure-structure interaction, and it is pointed out that this effect won't be negligible in the case buildings are massive and heavy. In order to clarify and estimate the effect, NUPEC under the authpiece of MITI, started this project in April 1994 [4],[5]. The project consists of two sub-tests, i.e., field and laboratory tests. The field test is carried out in the Higashidori site in Aomori Prefecture, located in the northern part of Honshu island of Japan. In the test, two types of adjacent structure models scaled down by about 1/10 are used. One is to examine dynamic cross interaction of the same two structures i.e., reactor buildings, and the other is to examine the interaction of different structures i.e., reactor building and turbine building. Also a single reactor building model with the same scale is constructed separately for comparison purpose.
The schematic drawing of the test site and building models is shown in Fig.1 and a picture of test models (the adjacent reactor building models) is given in Fig.2. The field test consists of shaker test and earthquake observation. These tests are carried out under two different conditions, with and



Adjacent Reactor Building Models

Reactor and Turbine Building Models

Fig.1
Test Site (Higashidori, Aomori Prefecture) and Building Models for the Field Test.

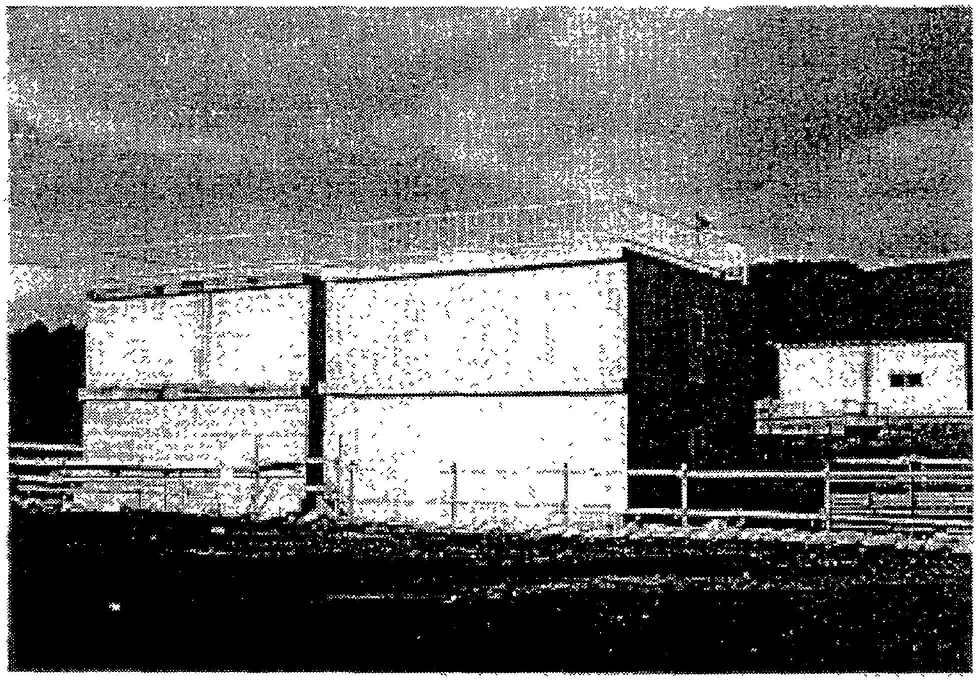


Fig.2 North-East View of the Adjacent Reactor Building Models

without embodiment of buildings. Laboratory test is planned to compensate a case of limited field test. The model consists of several small reactor building models made of aluminum, which have a 30centi-meters square cross-section in dimension and 38cm in height (1/260 scale), turbine building models with the same scale and a soil model which is made of silicon rubber (2.8m-in diameter and 1.0m-in height). The model is mounted on a shaking table and dynamic motions, including simulated design earthquake motions, are applied to study the effect of adjacent structures on earthquake response characteristics of a reactor building. The test will be completed by the end of March 2002.

1.2. MODEL TEST OF MULTI-AXES LOADING OF RC SHEAR WALLS

The cbjective of this project is to study dynamic response characteristics of NPP reactor building under three dimensional loading conditions which occur during a major earthquake. The current design analysis deals with three directional earthquake components independently. Then only the in-plane force on RC shear wall is evaluated as earthquake load. In order to know the ultimate strength of the RC shear walls, the effect of out-of-plane force on the strength should properly be evaluated as well. The motive of the study is to learn the ultimate strength of RC shear wall under three dimensional earthquake loads. From this standpoint, NUPEC started this project under the sponsorship of MITI in April 1994. The project consists of two sub-tests, i.e., static-cyclic and dynamic tests [4],[6]. The static-cyclic test includes following four sub-tests ; (1) an element test in which both shear-force and normal-force are applied to RC plates, (2) an element test in which simultaneous in-plane and out-of-plane loads are applied, (3) lateral diagonal loading test of box type shear walls and (4) simultaneous multi-axes loading test which is performed by applying simultaneous orthogonal horizontal and vertical loads and/or simultaneous orthogonal horizontal loads. The schematic drawings of the test concept are shown in Fig.3 and typical test view of the simultaneous horizontal and vertical loading test is shown in Fig.4. The dynamic loading test will be started from 1998. The test will be carried out to confirm restoring force characteristics of RC shear wall obtained by the static-cyclic loading test can stand on The test will be completed by the end of March 2004.

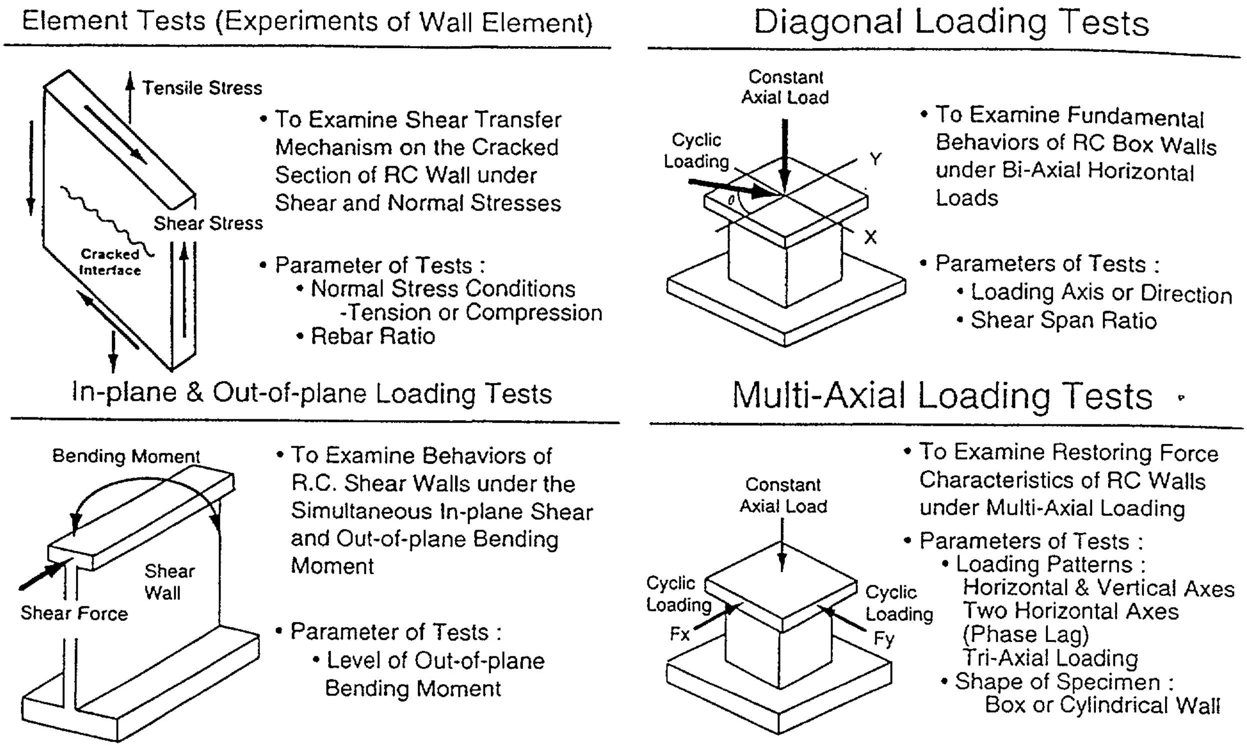


Fig.3 Basic Concept of the Static-Cyclic Test.

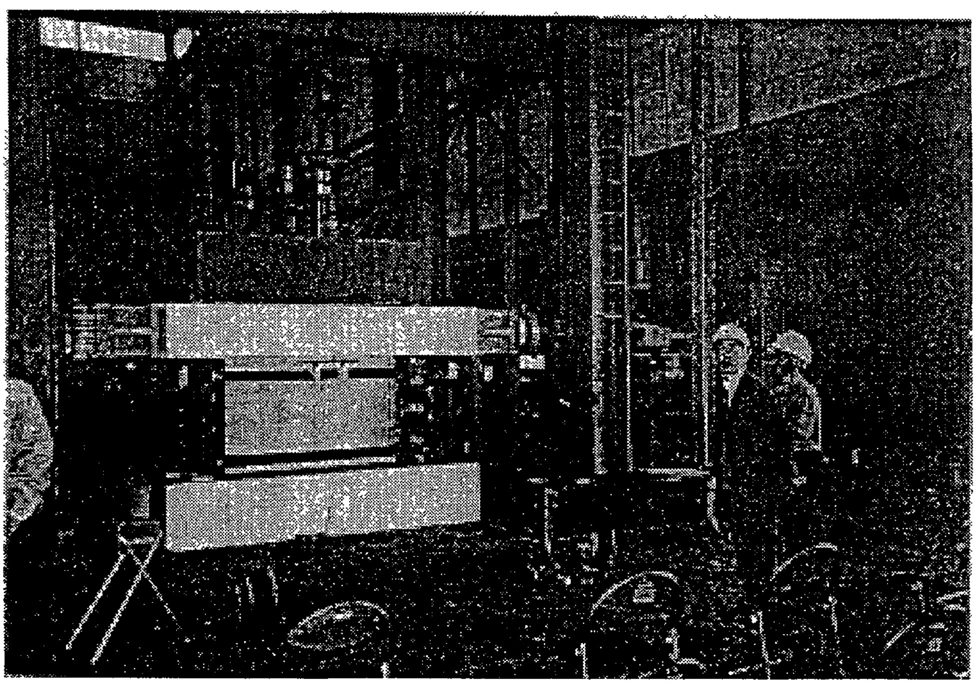


Fig.4
A Typical Test View of the Simultaneous Horizontal and Vertical Loading Test.

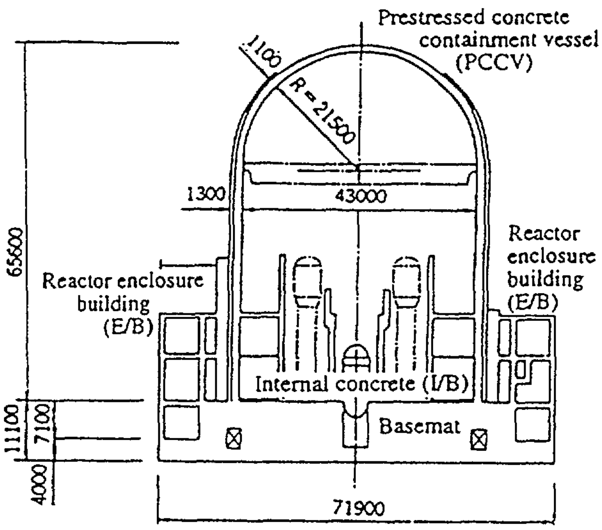
1.3. SEISMIC PROVING TEST OF CONCRETE REACTOR CONTAINMENT VESSELS

This project is planned as part of the seismic proving tests of NPP facilities which has been carried out by NUPEC using the large-scale, high-performance shaking table at Tadotsu Engineering Laboratory [7]. The project consists of two tests on concrete containment vessels, i.e., a PWR Prestressed Concrete Containment Vessel (PCCV) and a BWR Reinforced Concrete Containment Vessel (RCCV). The tests are to prove structural and functional integrity of a PCCV and a RCCV for the design earthquake S1 combined with design pressure, and for the design earthquake S2 unpressurized. In addition, the ultimate capacities of a PCCV and a RCCV to withstand earthquakes will be investigated.

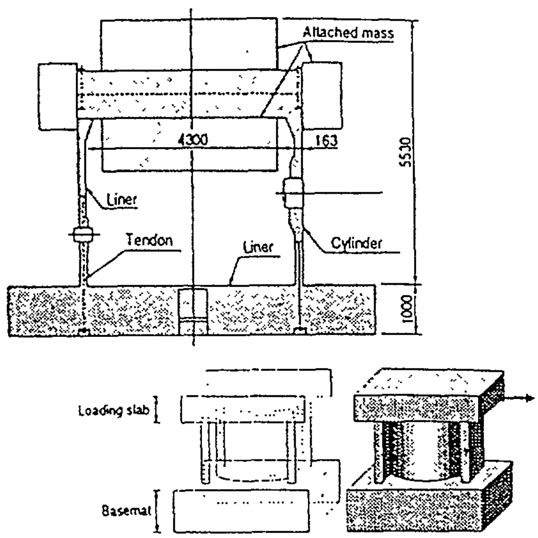
1.3.1 OUTLINE OF PCCV TEST PLAN

The test model is determined through a detailed investigation given to a concrete cylinder shown in Fig.5 [8]. The scale of the PCCV model is 1/10 of 1,100MWe PWR plant in Japan. In this model concrete dome is omitted because the seismic load on this portion is not critical. Thus it is replaced by a flat concrete slab, therefore the inverted U-shaped tendons which are applied to actual PWR plant are replaced by one through vertical tendons. A total of 434tons of additional lead masses are attached above, below and circumference of the top slab, to lower the natural frequency of the model and to compensate the seismic load reduction due to application of a scale model. The main test body measures 4.63m in outer diameter and is 6.53m in height including the basemat and additional lead mass. The model has a steel liner plate inside. The liner is 1.6mm-thick which is 1/4 scale to an actual liner plate, and this thickness is the minimum fabrication limit with proper tolerance. Thus the scale of the pitch of the liner anchors is also determined as 1/4. However, the depth of liner anchor is determined as 1/8 because of short intervals of rebars and/or tendons of PCCV. The design pressure for the test model is 4.0kg/cm^2, which is the same as that of an actual PCCV. The scales of the test model are summarized in Table 1. Total weight of the test model is 760tons. A picture of the test model is shown in Fig.6 [9]. The horizontal input earthquake motions of S_1 and S_2 to be used for the test were generated to fitting the design response spectra which are determined by enveloping the design earthquake ground motions of an actual PWR plant and the results of the preceding study on the design earthquake ground motions performed under the sponsorship of MITI. The vertical earthquake ground motions for the proving test is generated by fitting vertical design response spectra which are determined by multiplying the horizontal spectra by the factor of 0.5. The factor is the maximum value of the horizontal-to-vertical spectral component conversion coefficients which are proposed as the result of the project performed by NUPEC.

The test is carryng out from February to June in 1997 which consists of following four sub-tests; (1) preliminary test to investigate the dynamic characteristics of the model at a low level of acceleration, (2) verification test on design analysis method to check for the basic response characteristics by applying sinusoidal and/or simulated earthquake ground motions, (3) proving test which confirm the structural and functional integrity for design earthquakes of S_1 and S_2, and (4) seismic margin test which is carried out to comprehend seismic margins of the CCV to the design earthquakes.



Section of Reactor Building of a PWR



Test Model

Fig.5 Outline of Actual PCCV and Scaled Test Model.

Table 1 Scale of The Test Model

	Concrete	Liner		
Scale	1/10	Plate Thickness	Anchor Pitch	Anchor Depth
		1/4	1/4	1/8

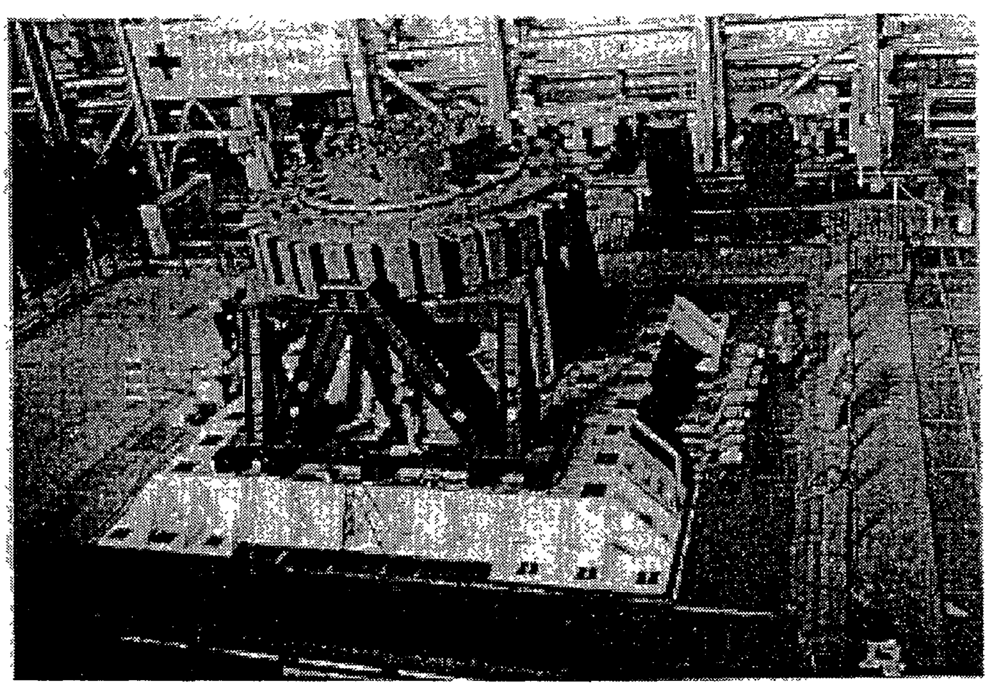
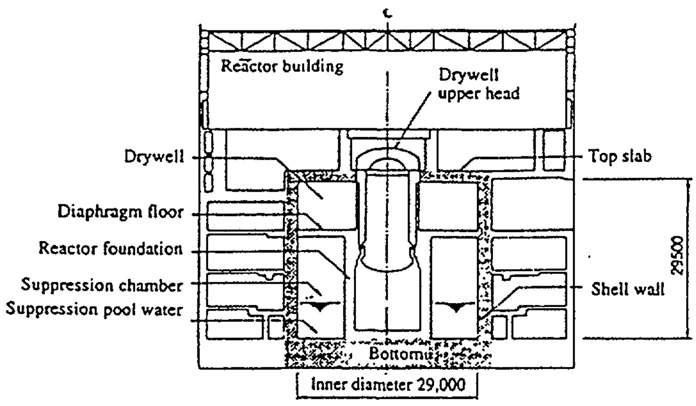


Fig.6
A Bird's-Eye View of the PCCV Test Model Installed on the Shaking Table at Tadotsu Engineering Laboratory, NUPEC.

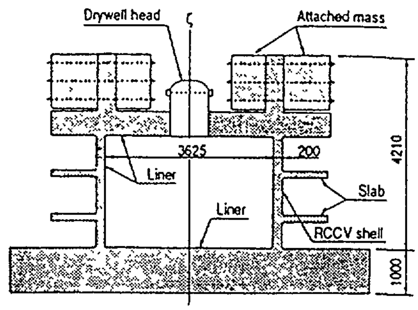
1.3.2 OUTLINE OF RCCV TEST PLAN

The aim of the RCCV proving test is to confirm dynamic characteristics, structural integrity and functional toughness against leakage of contained gaseous materials during and after the design earthquakes [9]. After the proving test, the seismic margin test will be carried out to learn dynamic behavior and seismic design margin to the design earthquakes by applying the large earthquake motions which exceed the design earthquake motions of S_2, as large as possible up to the limit of shaking table performance. The RCCV test model consists of a cylindrical shell wall, a topslab, a bottom slab, steel liner and additional lead masses. A drawing of the designed test model is shown in Fig.7 which is scaled 1/8 of an actual RCCV of a 1,350 MWe ABWR plant. As for dimensions, the test model is 5.63m in diameter (main test body) and 5.2m-high including basemat and additional lead masses. The RCCV shell wall is 20cm-thick whose scale is 1/10 of the actual RCCV. The scale of the pitch and depth of the liner anchors are determined 1/4 of those of the actual RCCV. Total weight of the model is 580tons including additional lead masses of 280tons. The additional lead masses lower the natural frequency of the model and compensate the seismic load as described in the previous section of PCCV. Fabrication of the model will be completed by the end of March 1998. Then the test will be started in April 1998 and completed by the end of July 1998. Detailed test plan is currently under examination.



Note: The shaded portion indicates the boundary of the reactor containment vessel.

Section of Reactor Building of an ABWR



Test Model

Fig.7 An Outline of Actual RCCV and a Scaled Test Model in Planning.

1.4. APPLICATION STUDY ON SEISMIC BASE ISOLATION TO NPP BUILDING

In recent years, there has been growing hopes to apply the seismic base isolation system to an NPP. Particularly, its application is expected for FBR (Fast Breeder Reactors) because reactor coolant temperature of an operating FBR is designed over 500℃ so that excessive seismic design for equipment and piping makes their thermal design difficult and that deteriorates economics of overall plant construction. Thus the particular effort to realize seismically isolated NPP was made in the feasibility study on a commercial demonstration plant of FBR [4]. Although seismically isolated buildings have been already constructed by many private construction companies in Japan, further verification studies are required with respect to the safety and integrity of the seismic isolation technology when it is applied to an NPP. From this standpoint, two major studies on seismic base isolation technology are carried out. One is a large scale project called "Verification Tests on FBR Seismic Isolation Systems". It was a MITI-founded project entrusted to CRIEPI (Central Research Institute of Electrical Power Industry) and was carried out from 1987 to 1994. In this study, the following items were investigated ;

(1) assessment on characteristics of large seismic isolation elements,

(2) assessment on dynamic vibration characteristics of seismic isolation system,

(3) assessment and study on appropriate seismic isolation structures,

(4) setting of design basis earthquake ground motions for seismic isolation NPP building,

(5) assessment on reliability of seismic isolation systems,

(6) development of seismic isolation design procedures.

As the result of this study, a draft guideline entitled "Design and Technical Guidelines on Seismic Isolation" was proposed [4]. Authorization of the draft is currently under discussion. Also the modification of the current standard earthquake design spectrum so called "The Ohsaki Spectrum" is investigated to raise the lower frequency component less than 2.0Hz by using fault rupture models of various kinds. Because the natural frequency of seismically isolated building tends to be designed around 0.5Hz. From the viewpoint of the frequency components lower than 2.0Hz, it is pointed out that the standard design spectrum proves smaller than that of recorded obtained by actual major earthquake such as Hyogoken Nanbu earthquake as shown in Fig.8 [10].

The other study is a FBR research common to electric power companies in Japan. In the study, a conceptual design of FBR as shown in Fig.9 [11] as well as an estimation of ultimate behavior of seismically isolated buildings and seismic fragility of the isolation systems are made by three-dimensional seismic response analyses designed to consider the rupture phenomenon of the isolation system. Figure 10 shows typical fragility curves and a typical rupture strength distribution of a base isolation system consisted of 367 isolators [12]. The results were obtained by 3-D (3dimensional) and/or 2-D earthquake response analyses. The study result demonstrated appreciable seismic safety margin against the design-base earthquake S_2 and proved that the rocking response of the building has a significant influence on the ultimate behavior rather than on the torsional response of the isolation systems.

Like FBR, application studies of the seismic isolation system to light water reactors have been carried out. The studies are focused on the plant construction cost reduction and the standardization of plant seismic design [13].

2. STUDIES ON UPGRADING OF SEISMIC DESIGN ANALYSES

An upgraded modeling technique usable in seismic design analyses of NPP structures is essential in streamlining seismic design. Recent remarkable progress in computer performance enables us to use sophisticated complex nonlinear analysis models for reasonable cost in curring in simulation of earthquake response of NPP structures when their site is hit by a strong earthquake. From this standpoint, many analytical studies are under way to find rational modeling for important structures. In this section, following two studies are described as example of this category;

(1) Comprehensive Applicability Studies of Test Data to Actual Plant Model,

(2) Seismic Design in Consideration of Vertical Seismic Ground Motion.

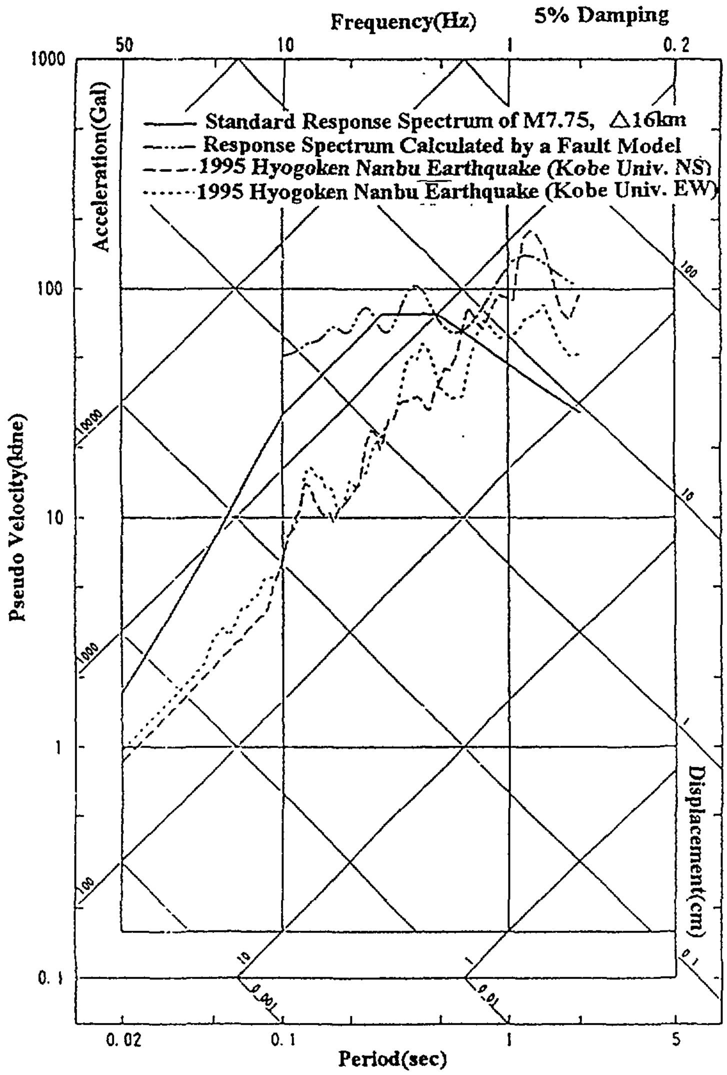
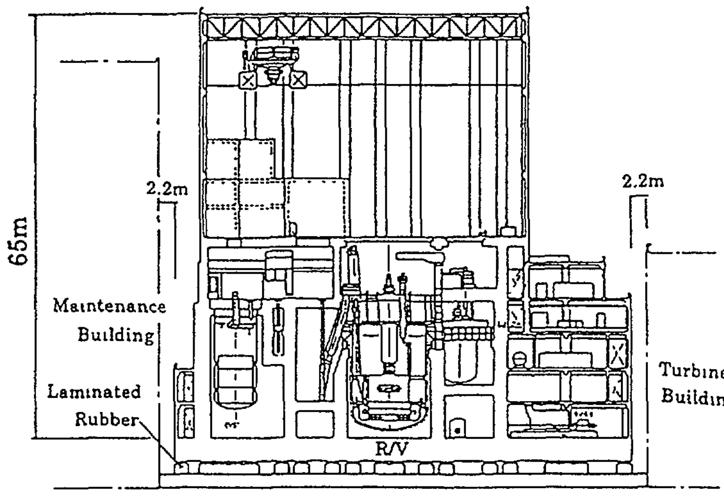


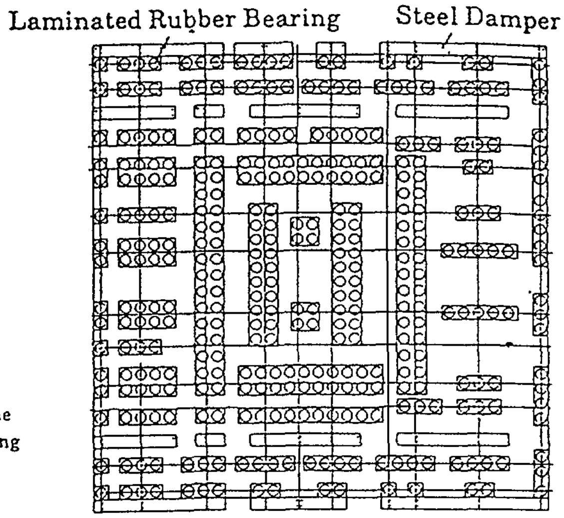
Fig.8 Comparison of a Standard Design Response Spectrum with the Response Spectra of Observed Earthquake Records and the Response Spectrum calculated using a Fault Model.

2.1.COMPREHENSIVE APPLICABILITY STUDIES OF TEST DATA TO ACTUAL PLANT MODEL

The project is designed for a comprehensive review and compilation of the test data and results obtained from preceding six tests to improve seismic safety analysis codes usable in an actual plant analytical model for the seismic design [14]. These tests are (1) model test on restoring force characteristics of reactor building, (2) model test on dynamic interaction between reactor building and soil, (3) model test on basemat uplift of reactor building, (4) model test on enbedment effect on reactor building, (5) evaluation test on inelastic seismic response of reactor buildings and (6) experimental evaluation of floor response spectra [15]. Conceptual drawings of these tests are shown in Fig.11. Also an outline of this program is shown in Fig.12. Result of this study is expected to produce a sophisticated analytical model (one of the best estimate models for the present) for seismic design analysis which contribute to upgrading of seismic design analyses. Also the test data obtained in preceding test projects will be compiled, updated and stored so that they can be referred to in the future effort to furthen upgrade plant seismic design analyses.

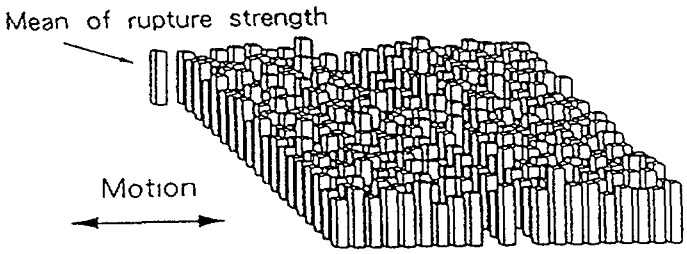


Section of FBR Reactor Building

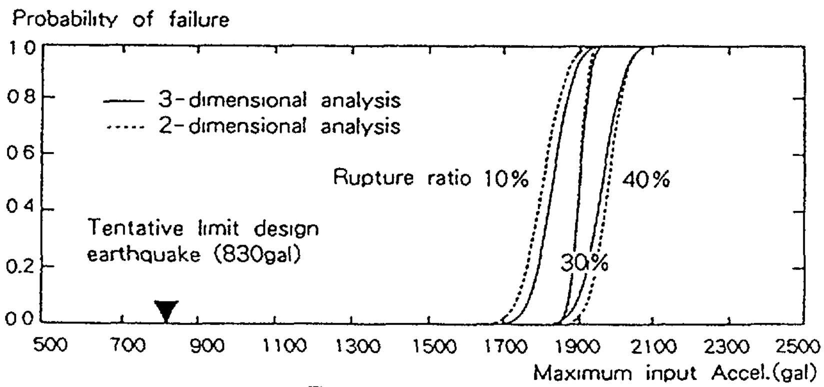


Arrangement of Seismic Isolator Devices

Fig.9
Seismic Base-Isolated FBR Reactor Building and Arrangement of Seismic Isolator Devices. (Reproduced from Ref.[9])



Rupture Strength Distribution



Fragility Curves

Fig.10
Typical Fragility Curves and Rupture Strength Distribution of Seismic Isolator Devices. (Reproduced from Ref.[10])

2.2. SEISMIC DESIGN IN CONSIDERATION OF VERTICAL SEISMIC GROUND MOTION

In the current technical guideline on seismic designs of NPP, vertical seismic load is statically dealt with. However, to upgrade the seismic design methodology, a design in which dynamic seismic loads in both horizontal and vertical directions are considered simultaneously could be required first. For this purpose, the design earthquake ground motions in the vertical direction must be determined.

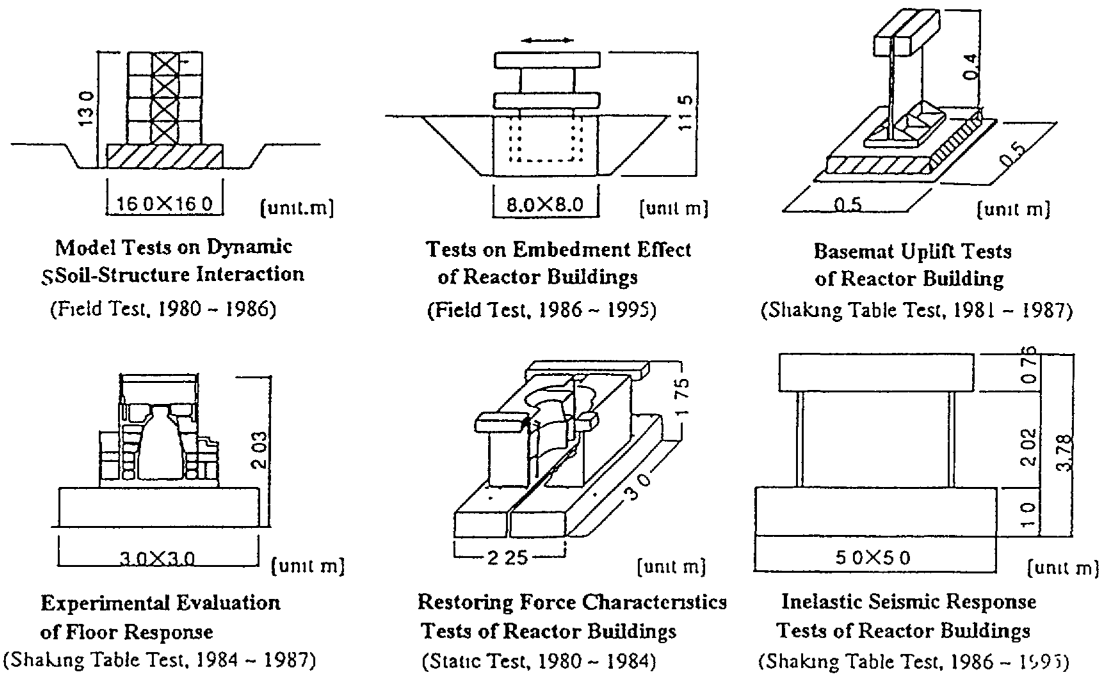


Fig.11
Basic Concept of the Preceding Tests Performed to Improve Seismic Safety Analysis Codes.

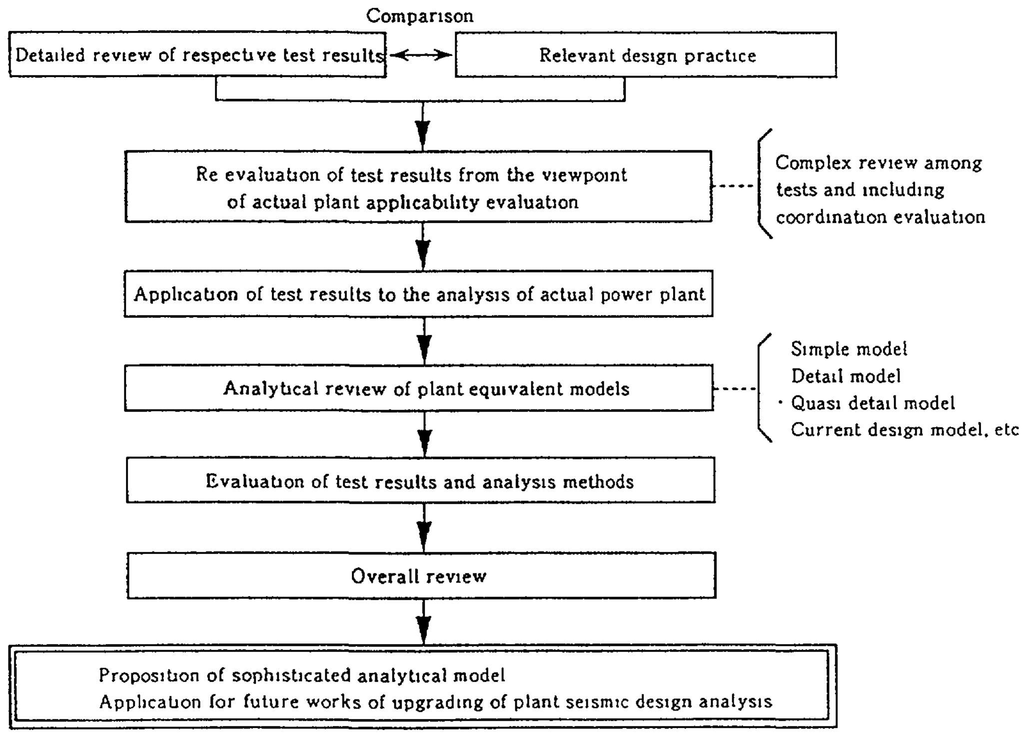


Fig.12
Outline of Comprehensive Applicability Studies of Test Data to Actual Plant Model.

The vertical design earthquake ground motion is defined in the form of response spectrum as it defined in the horizontal direction. In this project, the vertical-to-horizontal response spectral ratios (see Fig.13) of major earthquake records observed on rock field were investigated. By using the results, a conversion coefficient of horizontal-to-vertical component of design earthquake response

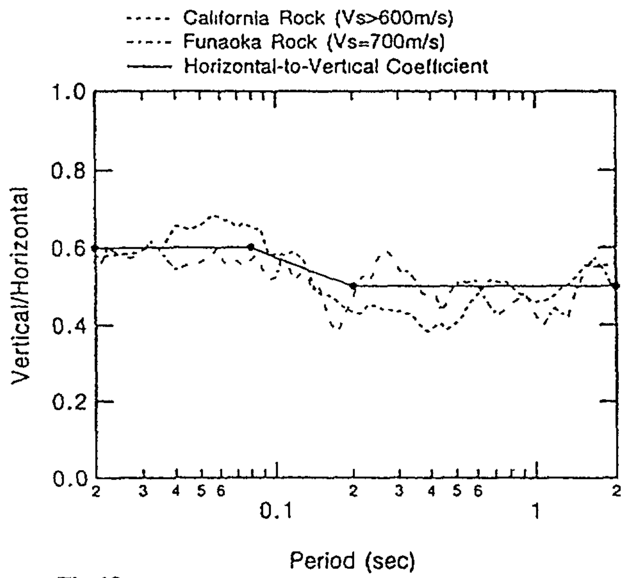


Fig.13
Comparison of the Proposed Vertical-to-Horizontal Response Spectral Ratios with those of Recorded Major Earthquakes.

spectrum was proposed [16]. At the same time, a rational synthesizing method of design earthquake motions was proposed [17]. Then to demonstrate this method, typical five design earthquake ground motions which have both horizontal and vertical components were generated. Applicability of the conversion coefficient of horizontal-to-vertical response spectral component to the earthquake record of the 1995 Hyogo-ken Nanbu Earthquake (Kobe Earthquake) and its aftershocks were investigated and confirmed enough applicability which indicates the reliability of the conversion coefficient.

The methodology and models of earthquake response analysis of structures which can simultaneously deal with both horizontal and vertical components of input earthquake ground motion were also studied. Based on the study results, prototypical analytical reactor building models for PWR and BWR plants were proposed. Figure 14 shows an example of BWR reactor building model. Then simultaneous horizontal and vertical earthquake response analyses were demonstrated by applying the design earthquake ground motions synthesized in this project.

The project will be expanded to develop the methodology which can deal with equipment earthquake response in the horizontal and vertical directions simultaneously.

Then a rational seismic design in which ground motions of both horizontal and vertical directions are considered will be proposed.

3. REEVALUATION OF EXISTING PLANTS FOR SEISMIC LOADING

The latest Japanese Examination Guide for Seismic Design of Nuclear Power Facilities was issued by NSC (Nuclear Safety Commission) in 1978 [18] and was partly revised in 1981. At that time, a total of 28 NPPs were already constructed and/or permitted to be constructed. Although those NPPs were confirmed of their seismic safety because they met the strict technical standards which were based on nearly the same concept with the Examination Guide, the NPP owner utilities made timely and careful reevaluations of the plant pursuant to the Examination Guide. The evaluation was made as part of their

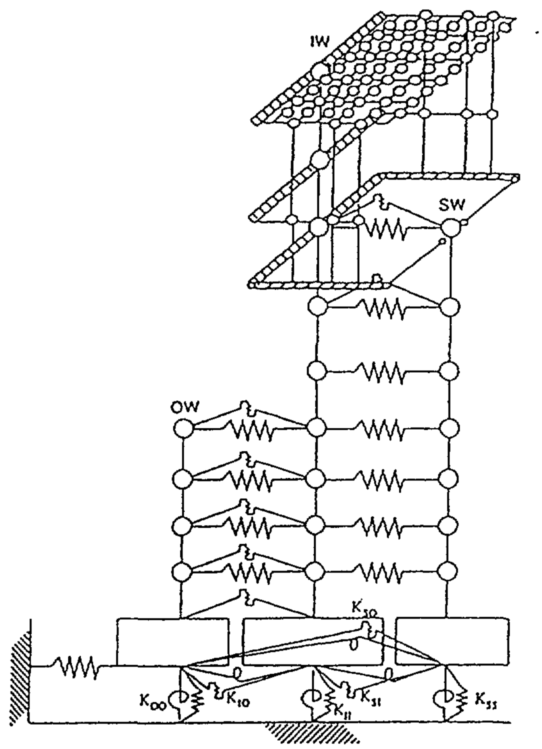


Fig.14
An Example of NPP Reactor Building Model (BWR) which can Calculate Horizontal and Vertical Earthquake Responses Simultaneously.

voluntary efforts to commit to security measures. Methodologies employed in their reevaluation studies were the same as those of current practice, expressed in the Technical Guideline for Seismic Design of NPP, JEAG-4601, issued by JEA(Japan Electric Association) [19]. Generally, every facility of Japanese NPPs has enough margin in its seismic design. These margins helped to produce favorable reevaluation results of older plants. The utilities submitted to MITI plant-by-plant calculation reports on their reevaluation results of the older plants. Assisted by a committee of NSC, MITI examined the reports and the examination was almost completed by late 1994 concluding that there would be no problem for further remodification [4],[20].

Although it is not a case of nuclear facilities, we introduce a seismic reevaluation of high pressure gas facilities including liquefied gas facilities as for a typical seismic reevaluation example in Japan. Those facilities have been under the control of MITI Notice #515, which was issued in 1981. Then we had been working for the remodification of the details of the Notice, and the new version was issued in March 1997. Those critical facilities are actually under the control of local governments. Some of them have been asking the reevaluation of the seismic design of related facilities in the case as follows ;

i) If the seismological survey shows the possibility of higher level of ground motions than the design basis ground motion specified in the MITI Notice #515,

ii) If the facility had been designed before 1981, and its design level was lower than the required design basis ground motion currently.

To back up for such requirements, Kanagawa-Prefecture have been developing techniques for reevaluation and reinforcement since 1984, and recently they issued the summary of their activity as a bounded paper volume consisted of six reports with examples. The documents related to those activities have been used as the text of the lecture by H.SHIBTA, one of the authors, in Seismic Design Course, International Program for Safety Management at NPPs which is held by Association for Japan Electric Power Information every year. This short course is planned mainly for specialists in Eastern Europe, Russia and China. Those consist of simplified elasto-plastic design and new methods for reinforcing existing facilities.

In addition to those, the evaluation of anchor-bolts is important. Some testings were done and summarizing them as design techniques in JEAG 4601-1991, "the seismic design guideline of nuclear power plants", but unfortunately, this part of JEAG-4601 has not translated into English. And in the 1997 new version Notice, we introduce L_2 ground motion in addition to L_1, the previous design basis ground motion. For L_2 ground motion, their allowable limit is similar to those for safety-related components of NPPs under the condition of level IV (or D). Again, its design procedure is a simplified elasto-plastic design. A factor, which is a function of ductility factor, has been employed, and this technique had been developed for the Building Code in Japan for 1981, and widely used for the design check of ordinary buildings.

4. INVESTIGATION TO EVALUATE AGING EFFECT ON NPP SEISMIC SAFETY

The share of power supply by NPPs in Japan amounts to 42,375MWe by August, 1997, over 30% of the total Japanese power supply. Japanese NPPs have been being kept high operation rate over 70% per year per each plant and the frequency of unexpected outage has been kept very small since 1983, one of the highest level in the world. However, the oldest three NPPs of light water reactors gradually advancing their ages, over 26years. Standing on this background, a proposal which pointing out the necessity of the investigations on countermeasure to aged NPPs was made as an urgent issue in the 1994 interim report of nuclear section of the synthetic energy investigation committee in Japan [23].

The investigations were carried out based on this proposal [24]. This investigations consist of ① analyses of NPP operating experiences (analyses of trouble experiences), ②confirmation of a meanings of the investigations on aged NPPs of their safety, ③study on technical issues for aged equipment and structures of NPPs. Figure 15 shows a relationship between a number of troubles per year and NPP operating experiences. The figure shows a trend that a number of troubles per a plant gradually decrease with the increment of operation experiences and no particular trouble increment due to aging are observed. Taking into account this trend, the sub-committee suggests that aging of NPPs don't affect their safety immediately but it is important for considerably aged NPPs to investigate the reliability of their installations. Figure 16 shows a technical flow diagram of the nuclear safety evaluation procedure for an aged NPP.

As for the investigation on the item ③, the sub-committee classified NPP equipment and structures according to their nature into the easy group for inspection, repair and replacement and the difficult group. Then for the difficult group, detailed investigations were carried out. Figure 17 shows selected equipment and structures for a PWR and a BWR by this investigation. A 60years of the plant life was supposed in this study although it did not necessarily mean the intention of plant life extension to the period. Thus three (1 PWR and 2 BWRs) NPPs were investigated of their safety-related equipment and structures with regard to their degradation due to aging in material property and a high cycle fatigue due to vibration live load and/or thermal stress applied during a long-term plant operation.

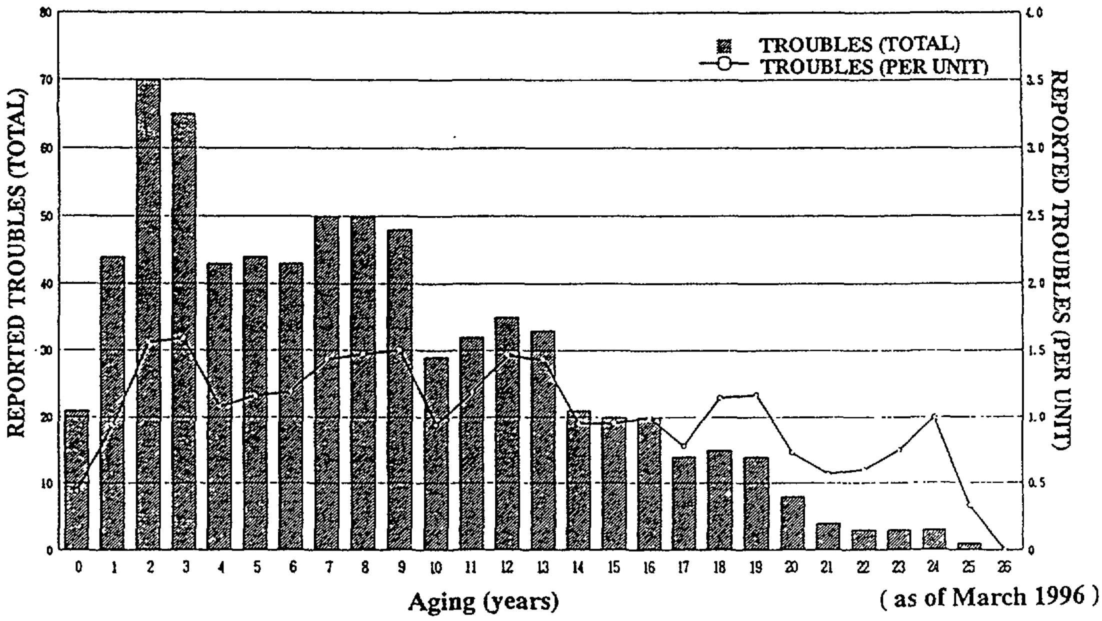


Fig.15
Transition of Annual Trouble Occurrences of Nuclear Power Plant in Japan

As the results of the studies, equipment and structures of the NPPs were confirmed to have enough safety margins against almost all aging phenomena. Moreover, action items for inspections and examinations needed to aged NPPs are extracted for future application.

Seismic safety evaluations were one of the most important items in this investigation. As for the items to be investigated for aged NPPs, followings were listed up to be a cause to deteriorate the mechanical material strength ; high cycle fatigue, neutron irradiation embrittlement, corrosion, stress corrosion crack and thermal embrittlement for metallic materials and the degradation in strength for reinforced concrete materials. It was concluded through the investigation that any severe seismic issues did not occur even for a considerably aged NPP unless it was confirmed that no explicit defect was observed during ordinary inspections and/or examination and a well organized preventive maintenance was carrying out.

5・CONCLUDING REMARKS

Japanese practice in seismic design of an NPP has been established. The methodology and procedure was reviewed after Hyogoken Nanbu Prefecture Earthquake (Kobe Earthquake) by the Japan Nuclear Safety Commission in the midst of glowing public concern over the seismic safety of NPPs after the earthquake and confirmed that NPPs in Japan have enough structural strength and functional raggedness against given design earthquakes. And yet, a wide range of studies have been continued to date by both public agencies and nuclear industries in hope to further upgrade NPP seismic design and confirmation of seismic safety of vintage NPPs. The studies described in this paper are just a few of such efforts. These studies are indispensable for making two originally incompatible subjects compatible, i.e., to reduce plant construction cost and increase seismic safety.

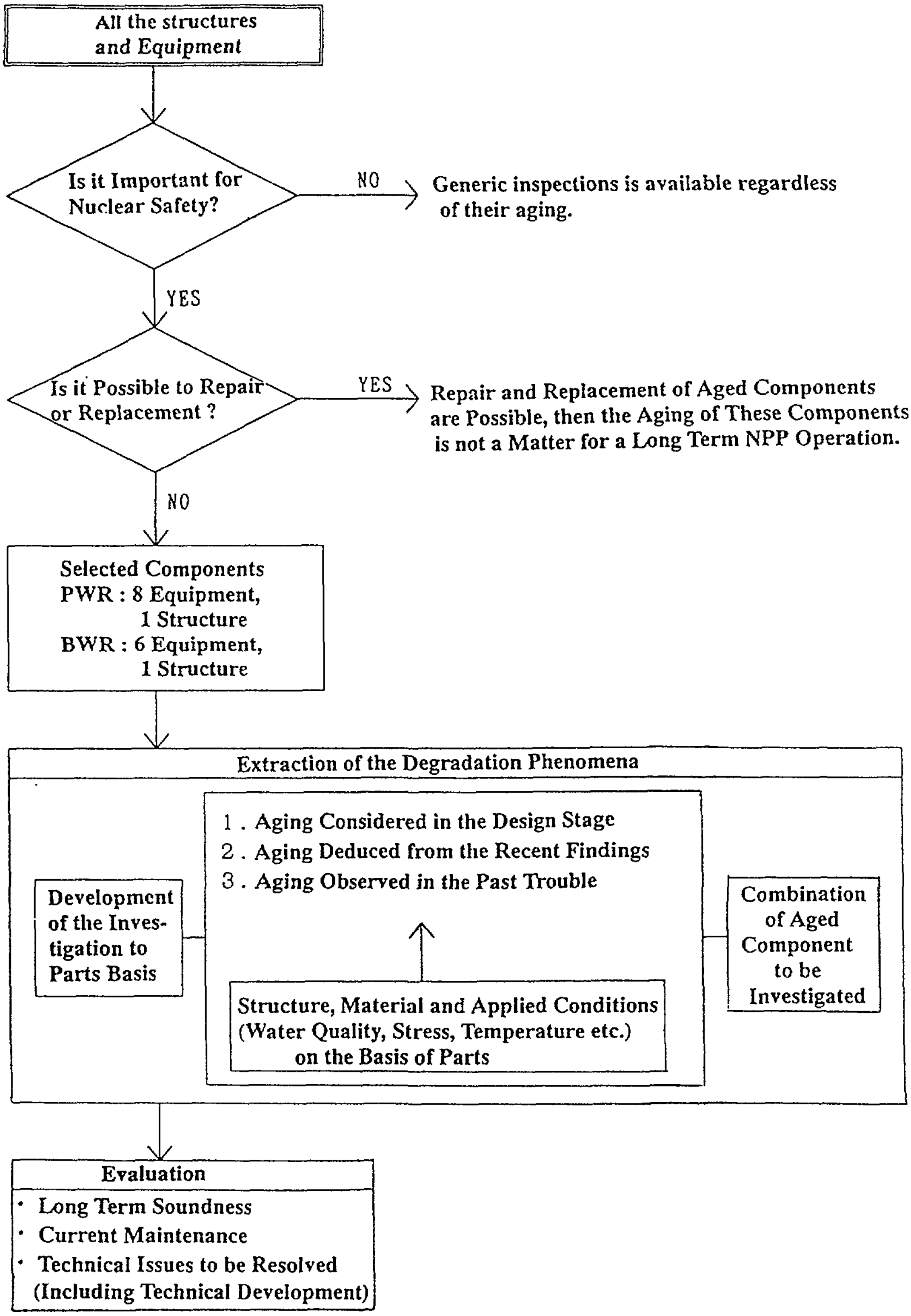
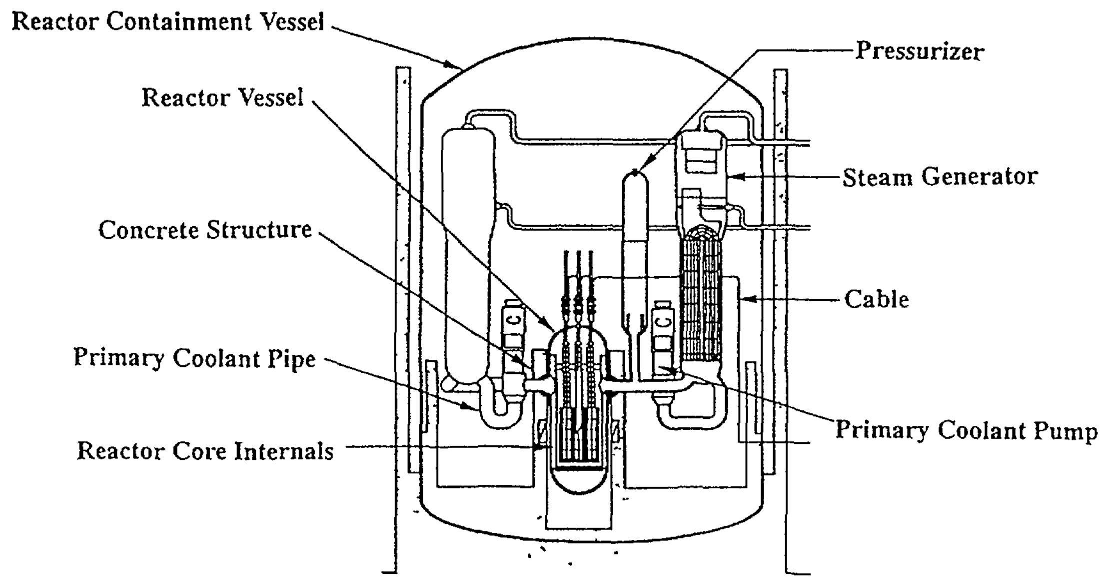
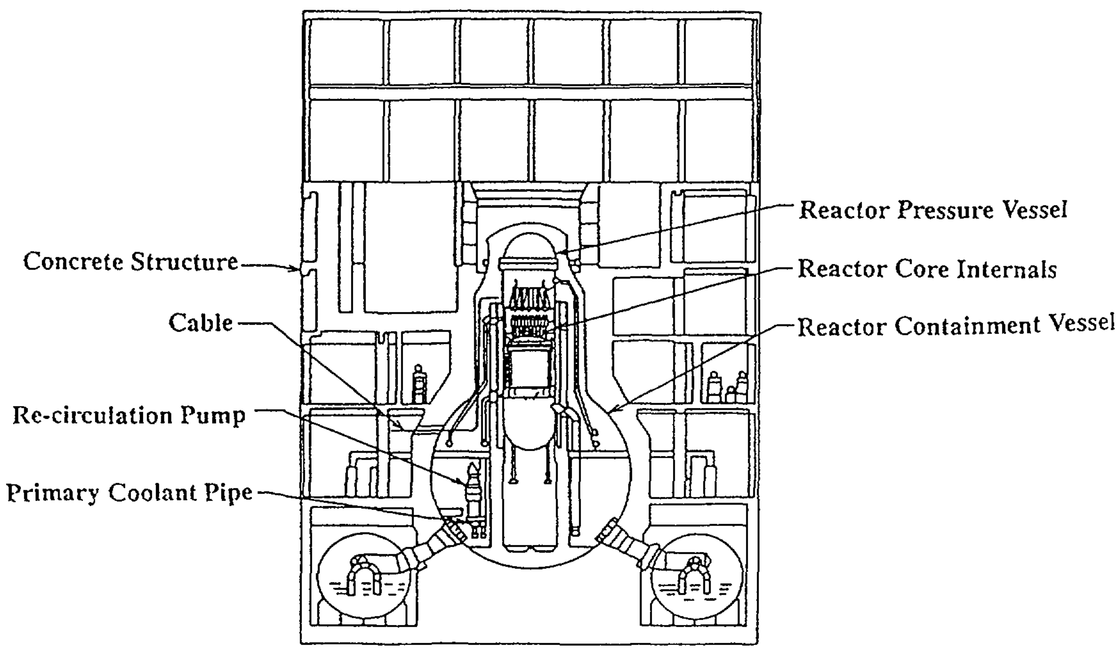


Fig.16
A Technical Procedure to Evaluate Degradation of Aged NPP Components



An Example of PWR



An Example of BWR

Fig.17 Selected Equipments and Structure for the Investigations

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SEISMIC ANALYSIS OF THE SAFETY RELATED PIPING AND PCLS OF THE WWER-440 NPP

XA0100523

A.M. BERKOVSKI, V.V. KOSTAREV, A.J. SCHUKIN
CKTI-VIBROSEISM,
Russian Federation

Z. BOIADJIEV
KNPP,
Bulgaria

M. KOSTOV
Risk Engineering Ltd,
Bulgaria

Abstract

This paper presents the results of seismic analysis of Safety Related Piping Systems of the typical WWER-440 NPP. The methodology of this analysis is based on WANO Terms of Reference and ASME BPVC. The different possibilities for seismic upgrading of Primary Coolant Loop System (PCLS) were considered. The first one is increasing of hydraulic snubber units and the second way is installation of limited number of High Viscous Dampers (HVD).

INTRODUCTION

One of the most important safety related systems of WWER-type NPPs are the piping providing Reactor Safe Shutdown function. Mainly these piping are located in Steam Generator (SG) and Main Cooling Pump (MCP) Boxes. On many of WWER plants these systems were designed according to former Soviet Union Standards and Rules, particularly by rather conservative PNAE Code. Nevertheless in some cases questions of seismic protection of the WWER units was out of the plant general design and criteria. That is why in the stream of the world community efforts to upgrade the nuclear safety of NPPs the great emphasis has been made for seismic reanalysis of WWER plants according to modern international practice.

This paper focuses on solving of seismic resistance problem for one of the old project of WWER-440-230 NPP. Initially in start-up period there were no any aseismic devices on PCLS and other safety related piping to withstand an earthquake and other extreme dynamic loads. The years after a number of hydraulic snubbers were installed on many of WWER units in spite of western practice to eliminate or reduce snubbers. This paper presents an accurate seismic analysis of safety related piping systems including PCLS according to modern international Standards on the base of accumulated engineering experience on other WWER NPPs.

METHODOLOGICAL BACKGRAUND FOR SEISMIC ANALYSIS

The main requirements for seismic analysis of equipment and piping of the WWER NPP are condensed in WANO developed "Terms of Reference and Technical Specification for Seismic Upgrading Design of KNPP Units 1 and 2" [1]. This document prescribes using of the Seismic Margin Assessment and ASME BPV Code, Section III approaches as methodological background [2] for seismic analysis of safety related piping located in Steam Generator and Main Cooling Pump Box. The Terms of Reference contains the following general recommendations for load combinations and allowable stress limits in seismic analysis, Table 1. The first column of this table shows the safety classes according to SRP 3.2.2 [3]. In the second column of the table are shown the load combinations (without brackets) strictly according to Terms of Reference and in brackets are pointed the load combinations in interpretation of SRP 3.9.3. The third column presents the formulas that were selected from ASME BPVC for implementation of Terms of Reference recommendations. The description of allowable stresses are shown in the fourth column of Table 1 and in the Table 2.

Table 1

Class	Load Combination	Formulas (NB-3650, NC-3650)	Allowable Stresses
1	DL+LL (P+DL+LL)	$B_1 \cdot \frac{P \cdot D_o}{2 \cdot t} + B_2 \cdot \frac{D_o}{2 \cdot I} \cdot M_i, (9)$	1.5 S_m
	DL+LL+EQi (P+DL+LL+EQi)	$B_1 \cdot \frac{P \cdot D_o}{2 \cdot t} + B_2 \cdot \frac{D_o}{2 \cdot I} \cdot M_i, (9)$	3.0 S_m
	T+EQm (P+DL+LL+EQm)	$C_1 \cdot \frac{P_o \cdot D_o}{2 \cdot t} + C_2 \cdot \frac{D_o}{2 \cdot I} \cdot M_i, (10)$	3.0 S_m
2	DL+LL (P+DL+LL)	$B_1 \cdot \frac{P \cdot D_o}{2 \cdot t_n} + B_2 \cdot \frac{M_A}{Z}, (8)$	1.5 S_h
	DL+LL+EQi (P+DL+LL+EQi)	$B_1 \cdot \frac{P_{max} \cdot D_o}{2 \cdot t_n} + B_2 \cdot \frac{M_A + M_B}{Z}, (9)$	3.0 S_h
	T+EQm	$\frac{i \cdot M_C}{Z}, (10)$	S_A

Table 2

Class	Description	Allowable Stresses
1	S_m	$\min \left\{S_T/3 ; 1.1S_T^T/3 ; S_Y / 1.5; S_Y^T/1.5\right\}$
2	S_c, S_h	$\min \left\{S_T/4 ; 1.1S_T^T/4 ; S_Y / 1.5; S_Y^T/1.5\right\}$
	S_A	$1.25 * S_c + 0.25 * S_h$

Two types of seismic excitation were stipulated for analysis: Review Level Earthquake (RLE) and Local Earthquake (LE) defined in terms of Response Spectra. ZPGA level for RLE was assumed as 0.16g. For the systems which were supported on different elevation levels of structure the envelope spectra has been developed according to Appendix N of ASME Code [4]. For evaluation of seismic capacity of considered systems two analytical approaches have been used: Response Spectrum Modal Analysis Method (RSMAM) and Time History Analysis (THA). In case of TH analysis the TH acceleration was generated from target Response Spectra following to demands of Appendix N of ASME Code (N-1210). The damping ratio for all piping systems was accepted as 0.05 [1].

The load combinations and allowable stresses for seismic capacity evaluation of piping and equipment supports were also defined on the basis of [1, 2] recommendations, Table 3.

Table 3

Element of Support	Load Combination	Failure Mode	Allowable Stresses
Steel Structure	DL+LL+T DL+LL+T+EQi+EQm	Plastic Collapse	Sall 1.6 Sall; 0.7 Su [1)]
Fixed Joints	DL+LL+T DL+LL+T+EQi+EQm	Plastic Collapse	0.5 Su 0.7 Su [1)]
Welded Joints	DL+LL+T DL+LL+T+EQi+EQm	Brittle	0.3 Su 0.42 Su [1)]
Springs of Hangers	DL+LL+T DL+LL+T+EQi+EQm	Limited compression	Pmax

[1)] The level of allowable stresses is defined according to Appendix F of ASME BPVC [2].

One of the most important features of the present methodology is possibility of using of inelastic demand-capacity ratio (ductility factor) that essentially decreases the conservatism of traditional Code (ASME as well as PNAE) approaches. The following recommended values of these inelastic coefficients were implemented to current analysis [5]:

- Distribution System Supports - Fu = 1.25
- Welded Joints of Piping Supports - Fu = 1.0
- Piping - Fu = 1.5

For the following elements of distribution systems the Fu coefficients were used conventionally in terms of device's operability under seismic excitation according to supplier's catalogues [6,7]:

- Springs of Hanger support - Fu = 1.0
- Hydraulic Snubbers - Fu = 1.0
- GERB Dampers (Nseism) - Fu = 1.0 (Nseism = Nnom x 1,7)
- CVS HV Dampers - Fu = 1.0

The strength analysis (seismic capacity of structure) was defined using the above pointed coefficients by the following formulas [16]:

- Stresses (reactions) from the inertial part of seismic load (EQi):

$$S_{Eid} = \frac{S_{Ei}}{F_U} \qquad (1)$$

- Stresses (reactions) from the seismic anchor movement (EQm):

$$S_{Emd} = F_U \cdot S_{Em} \qquad (2)$$

One of the serious obstacles for providing correct analysis of running plants is gathering of necessary authentic information and input data. The only way to solve this problem is realization of walkdown procedure for each system to be analyzed for defining the real terms of equipment, piping, system and their supports installation and operating. It is quite usual that in many cases the typical WWER-type NPPs shortcoming like insufficient lateral restraining is recognized.

The present seismic analysis covers the following systems and their elements: small and large bore piping, piping supports and piping nozzles (Reactor Pressure Vessel, Steam Generator, Pressurizer).

ANALYSIS RESULTS

The full finite-element analytical model of WWER-440 piping systems located in Steam Generator and Main Cooling Pump Box is shown on figure 1.

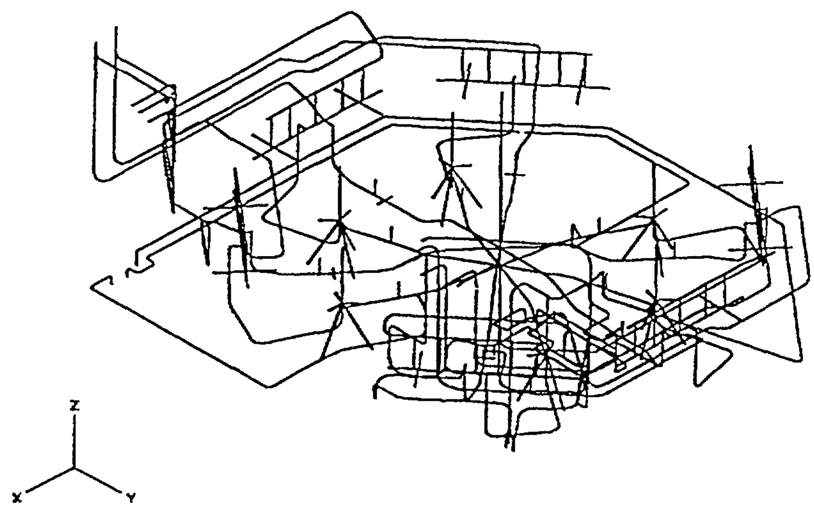

Figure 1 Complex Analytical Model of SG and MCP Box piping

This sketch includes detailed models practically of all large bore hot piping (with diameter more than 100 mm) and simplified models of Reactor Pressure Vessel, SG, MCP and connected equipment for all of six loops of PCLS.

The further consideration for more clear description of the main obtained results will be based on analysis of the first Primary Coolant Loop, Figure 2.

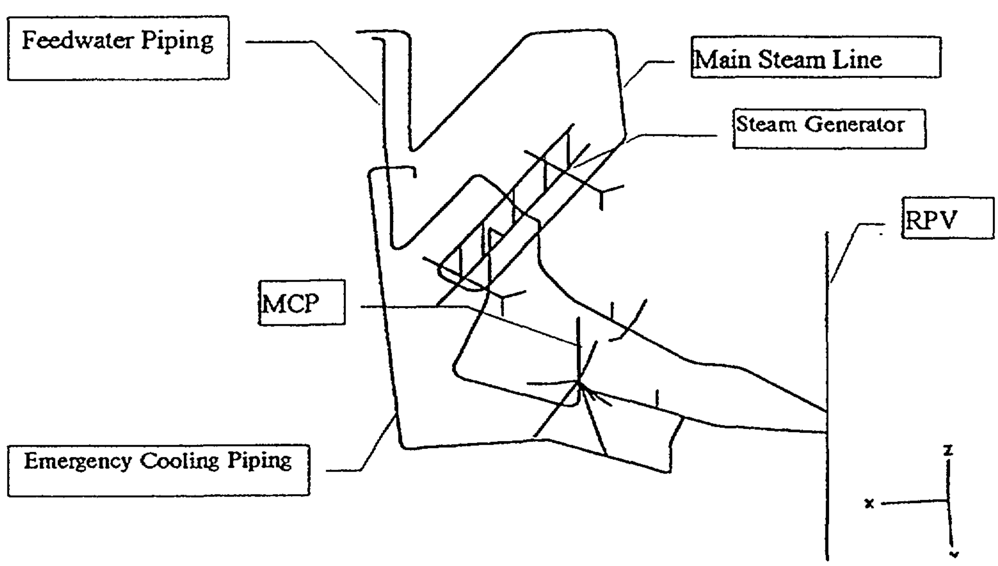

Figure 2 Analytical Model of the first PCLS

PCLS without seismic restraining (initial design)

The first natural frequency of PCLS is shown on figure 3.

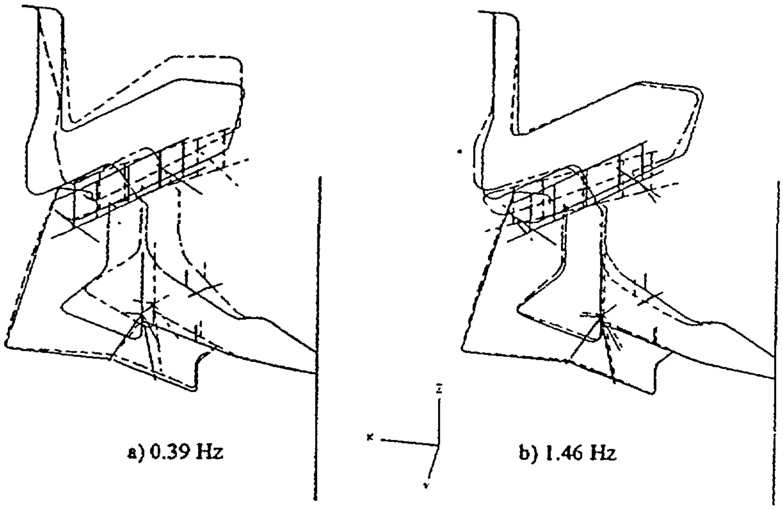

Figure 3 The first natural modes of PCLS without seismic restraining

The low level of PCLS natural frequencies leads to intensive seismic response of structure. The displacement of SG achieves more than 500 mm. Additionally the analysis of PCLS without seismic restraining shows that for many of piping elements (runs, bends and tee elements) the safety requirements are not satisfied even in case of using non-conservative ductility approach. That means that seismic upgrading of PCLS have to be performed obligatory to meet the demands earthquake protection and Terms of Reference. Thus the installation of hydraulic snubbers that was performed in eighties on a number of Ukrainian and East European WWER NPP Units is quite feasible and was in the stream of that and previous time experience.

PCLS with snubber restraining

Figure 4 demonstrates the principal location and types of hydraulic snubbers that usually are installed on PCLS.

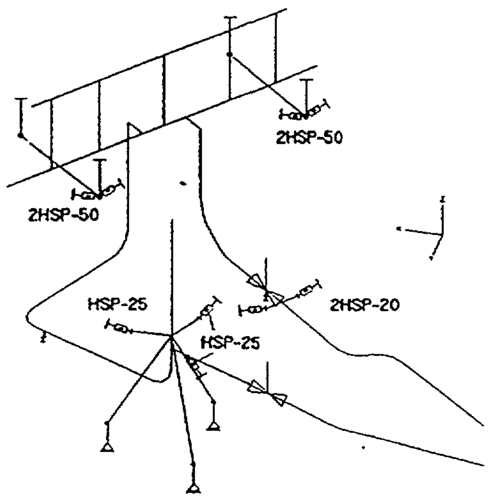

Figure 4 Snubber Location for Loop No 1 of PCLS

The accurate comprehensive non-linear TH analysis of this system has been performed to obtain the realistic dynamic response of the PCLS and snubber reactions. Dynamic characteristics of the snubbers based on their direct testing including specific velocity locking limits of the snubber's piston recommended by manufacturer [6] were involved in this analysis, Figure 5.

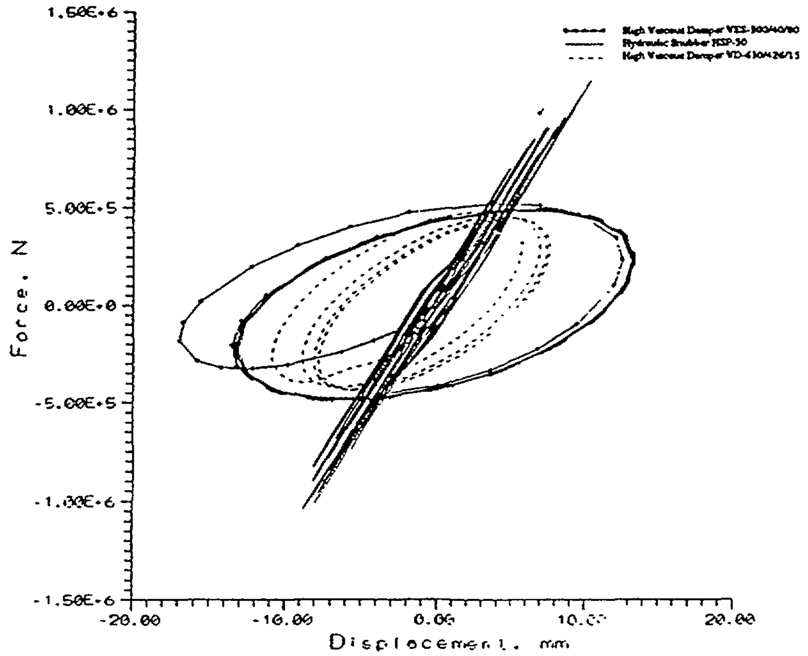

Figure 5 Dynamic characteristics of the ST Hydraulic Snubbers and High Viscous Dampers under sinusoidal 1 Hz excitation

This kind of analysis shows that there are not problems in seismic safety of PCLS and connected piping as itself. However the dynamic reaction of snubbers for some of devices exceeds the recommended capacity (limit load) of snubber for several times, Figure 6.

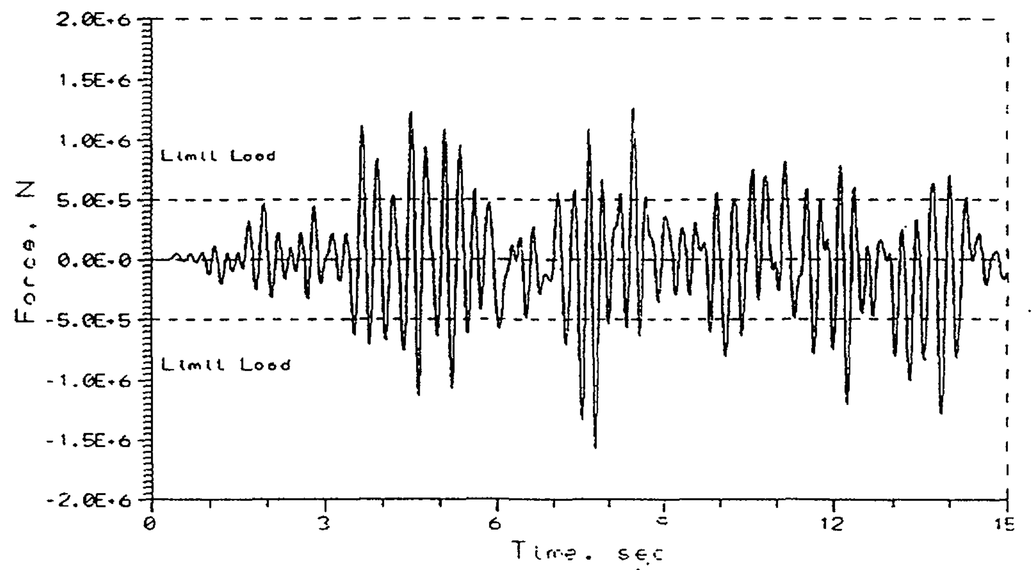

Figure 6 The TH seismic reaction force in overloaded snubber

That is why for meeting the requirements of seismic criteria the additional number of hydraulic snubbers have to be installed. The analysis shows that only double increasing of snubbers with the same load capacity under SG will solve practically the problem of PCLS seismic resistance. The reaction force of snubbers in this case do not exceed more than on 12% their nominal catalogue load capacity that seems to be acceptable. The total number of snubbers for one PC loop in this case increases from 9 to 13.

PCLS with High Viscous Dampers restraining (possible seismic upgrading)

In recent years the more reliable HVD technology has been widely implemented in seismic upgrading of WWER, PWR, BWR and other types of NPPs [7]. The dynamic characteristics, analytical model and significant advantages of these devices were investigated in literature in details [8-15], Figure 5. For purposes of this analysis the 4-parameters Maxwell Model of Viscous Damper that correctly reflects frequency-depended dynamic properties of HVD has been used [11].

Two variants of proposed location for case of HVD installation for PCLS is shown on Figure 7.

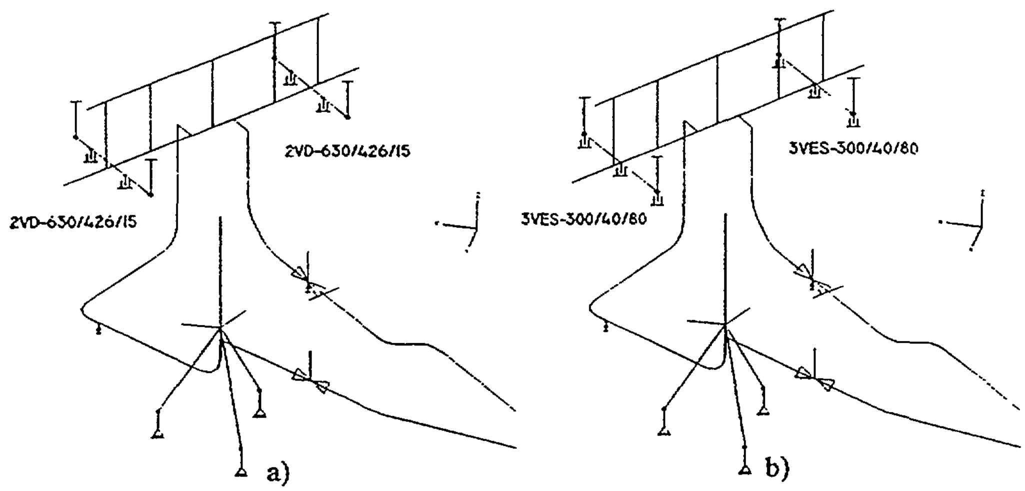

Figure 7 HVD location for Loop No 1 of PCLS

Time History Analysis of PCLS with HVD shows that four units of VD-630/426-15 is enough to provide sufficient seismic resistance of the Loop. In case of VES-300/40/80 installation this number increases up to 6 devices. In both cases stresses in piping, nozzles and supports are meet seismic criteria and requirements.

CONCLUSIONS

1. The accurate seismic analysis of WWER-440 NPP Safety Related Piping Systems and Nozzle Zones including PCLS has been performed to find out the way of possible seismic upgrading.
2. It was shown that for withstanding to earthquake with ZPGA more than 0.1g the application of special seismic devices to the WWER-440 Primary Loop is strictly recommended.
3. The analyses show that PCLS meets the seismic criteria and requirements in case of 13 snubbers versus 6 or 4 High Viscous Dampers depending on type of these devices. The additional benefit of HVD technology is high reliability of devices and low maintenance cost.

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