Tennessee Valley Authority, Post Office Box 2000, Spring City, Tennessee 37381

William J. Museler
Site Vice President, Watts Bar Nuclear Plant

JUN 15 1993

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555

Gentlemen:

In the Matter of the Application of	)	Docket Nos. 50-390
Tennessee Valley Authority	)	50-391

WATTS BAR NUCLEAR PLANT (WBN) - UNITS 1 AND 2 - CONTROL ROD DRIVE MECHANISM (CRDM) SEISMIC OPERABILITY (TAC NOS. M84249 AND M84250)

The WBN Supplemental Safety Evaluation Report (SSER)-10, Paragraph 4.2.3(2) discusses CRDM seismic operability at Watts Bar Nuclear Station Units 1 and 2. As documented in this report, "The staff has not yet reviewed all of the methods noted by the applicant for demonstrating its CRDM operability during and following seismic events." To facilitate the staff review required to close this issue, TVA has been requested to submit the following information to docket.

1. A detailed description of control rod operability including the various tests performed to address this issue by Westinghouse and the Japanese.

2. The basis for the recent CRDM statement revision in the Watts Bar Final Safety Analysis Report (FSAR).

Enclosed find Westinghouse letters WAT-D-9338, dated June 7, 1993, and WAT-D-8895, dated June 29, 1992, which provide the requested CRDM seismic operability information for Watts Bar Units 1 and 2. They discuss the generic testing programs performed both domestically and by the Japanese and their applicability to Watts Bar. They also discuss in detail the issue of CRDM operability and the basis for the recent FSAR revision.

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If any additional questions exist relative to this information, please contact P. L. Pace at (615)-365-1924.

Very truly yours,

W. J. Museler

William J. Museler

Enclosures
cc (Enclosures):

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Watts Bar Nuclear Plant
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ENCLOSURE 1
WATTS BAR NUCLEAR PLANT
UNITS 1 AND 2
CONTROL ROD DRIVE MECHANISM OPERABILITY

Westinghouse Letter
WAT-D-9338
June 7, 1993

T25 93 0608884

Westinghouse Electric Corporation
Energy Systems
Box 355
Pittsburgh Pennsylvania 15230-0355

WAT-D-9338
June 7, 1993

Ref. 1) WAT-D-8895

Mr. W. L. Elliott
Manager of Engineering
Tennessee Valley Authority
Watts Bar Nuclear Power Plant
IOB-1A, P.O. Box 2000
Spring City, TN 37381

Ans'd by Ltr # NAR

Attention: Steve Robertson

Tennessee Valley Authority
Watts Bar Nuclear Plant Units 1 & 2
CRDM Seismic Operability

Dear Mr. Elliott:

Westinghouse has confirmed that the information supplied via reference 1 concerning Control Rod Drive Mechanisms (CRDMs) is applicable to Watts Bar and should provide sufficient justification to demonstrate the seismic operability of the Watts Bar CRDMs.

If you have any questions on this matter, please contact this office.

Very truly yours,

J. W. Irons, Manager
TVA Watts Bar Project
Domestic Customer Projects

SER 1660

PCR				
GEB				
SL Robertson, IOB 1H			1	
DW Posey, IOB 1H			1	
W. Massie, FSB 2K			1	— per your request and OK
JG Adair, IOB 1F			1	

cc: W. L. Elliott, 1L
S. L. Robertson, 1L

ENCLOSURE 2

WATTS BAR NUCLEAR PLANT
UNITS 1 AND 2
CONTROL ROD DRIVE MECHANISM OPERABILITY

Westinghouse Letter
WAT-D-8895
June 29, 1992

T33 920708 962

Westinghouse Electric Corporation

Energy Systems

Nuclear Services Division

Box 355
Pittsburgh Pennsylvania 15230-0355

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~~NEB~~ DWPosey,	JOB 1H		1	
~~NEB~~ W. Massie,	FSB 2K		1	

WAT-D-8895
June 29, 1992

Mr. W. L. Elliott
Manager of Engineering
Tennessee Valley Authority
IOB-1A, P.O. Box 2000
Watts Bar Nuclear Power Plant
Spring City, TN 37381

Ref: 1. NSSS 92-041

Attention: Mr. S. L. Robertson

Ans'd by Ltr # NAR

TENNESSEE VALLEY AUTHORITY
WATTS BAR UNIT 1 AND 2
Control Rod Drive Mechanism (CRDM) Seismic Operability

Dear Mr. Elliott:

The following information is being provided in response to the 5 questions identified in reference 1 concerning Control Rod Drive Mechanism Seismic Operability.

1. History of Comanche Peak SQRT Audit:

   Given in Attachment I.

2. Testing that was done:

   A detailed description on Control Rod operability including the various tests performed to address this issue, by Westinghouse and the Japanese is contained in the Attachment to this letter.

3. Previous NRC Approval for Other plants:

   The same arguments were presented to the NRC during the licensing process of Comanche Peak Unit 1.

4. Basis for deleting statement in the FSAR:

   The statement was deleted because it incorrectly stated that the basis for demonstrating that control rods could be inserted during a seismic event was dynamic analysis.

5. Other related (W) Input:

Attachment III is a copy of a public paper," SEISMIC TEST AND ANALYSIS FOR PWR REACTOR CORE INTERNALS", as published in Volume 182 of the 1989 Pressure Vessels and Piping Conference sponsored by the American Society of Mechanical Engineers and the Japan Society of Mechanical Engineers. Attachment 4 is a copy of the public paper, "Control Behaviour in Earthquakes", as published in the April, 1990 edition on NUCLEAR ENGINEERING INTERNATIONAL.

Very truly yours,

Kuth Forlt for

J. W. Irons, Manager
Watts Bar Project
Domestic Customer Projects Department

/slf

Attachment

ATTACHMENT I

COMANCHE PEAK SQRT AUDIT-HISTORY

BACKGROUND AND PURPOSE

As part of the NRC's Equipment qualification Branch review of Comanche Peak, a Seismic Qualification Review Team (SQRT) audit was conducted to determine the acceptibility of seismic qualification. Comanche peak was the first Category I plant reviewed by the NRC. As such, the NRC evaluated Comanche Peak to the criteria in Standard Review Plan 3.10/3.11 (NUREG-0800) for both mechanical and electrical equipment. The Standard Review Plan requires compliance with IEEE 344-1975, IEEE 323-1974 and includes additional requirements for dynamic testing, aging of mechanical equipment, and sequence testing.

NRC AUDIT METHOD

The NRC and their consultants (Brookhaven Labs) pursued various objectives during this audit. These were:

1. Perform a walkdown of the equipment selected for the audit to determine the acceptibility of its installation (particularly mounting) relative to its seismic qualification.

2. Review of the seismic qualification documentation to determine the acceptibility of the seismic qualification methods and completeness and availability of the documentation.

3. A review of the site specific maintainance and surveillance program.

4. Develop conclusions concerning the acceptibility of the seismic qualification of equipment for Comanche Peak.

In pursuing these objectives the NRC focused on other related aspects of equipment qualification such as the acceptibility of seismic qualification methodology and computer codes which were reviewed by the Mechanical Engineering Branch.

NRC COMMENTS AND WESTINGHOUSE RESPONSE

As part of the above audit, one issue that was raised was definition of the basis for assuring that control rods could be inserted during or following a seismic event. Westinghouse explained how the CRDM pressure boundary is qualified by dynamic analysis and ASME Section III stress analysis. With respect to operability the following argument was presented:

1) Operability is defined as the ability to insert rods when desired.

2) The CRDM is a fail safe component since it is actuated by gravity only.

3) Rod drop time measurements during startup provide verification of operability.

4) Rod drop capability under abnormal conditions has been demonstrated by:

-Prototype flow tests over a range of temperatures and flows greater than 150% of nominal.

-Scram deflection tests on CRDM's, guide tubes and fuel assemblies.

- Furthermore, a Westinghouse licensee has performed dynamic tests on a prototypic CRDM which provide additional evidence of the ability to insert rods during seismic events.

Westinghouse informed the reviewers that the MHI data was proprietary and that we could not release the reports to them. (Since the time of the audit, some of the Japanes data has been released publically,, such as in the papers contained in`Attachment 4).

ATTACHMENT II

CONTROL ROD DRIVE SYSTEM OPERABILITY

INTRODUCTION

Control rod drive system operability is defined as the ability to insert the control rods when desired. The objective of this writeup is to summarize the basis for operability of the control rods for the safe shutdown earthquake. In a reactor, the active portion of the control rods which contains the neutron absorbing material is attached to a drive rod which can be axially positioned using a magnetically operated jacking mechanism. This mechanism is referred to as a control rod drive mechanism or CRDM. The CRDM's are mounted above the reactor vessel head and contain a long rod travel housing which allows space for axial movement of the drive rod and control rod. The CRDM's thus are part of the pressure boundary of the primary reactor coolant system and structural integrity of the CRDM must be assured to preclude leakage of coolant and to assure the ability to insert the control rods. Structural integrity is demonstrated by dynamic seismic analysis in combination with stress analysis to demonstrate that ASME code stress limits are met.

The CRDM's used in Westinghouse and Westinghouse licensee reactors rely on magnetic fields to withdraw the control rods from the core. When the electrical currents which generate these magnetic fields are removed, the control rod is released from the CRDM and is inserted into the core by the effect of gravitational acceleration. This design is referred to as "fail-safe" since control rod insertion does not depend on any external loads being applied other than the weight of the control rods. The moveable control rod assembly consists of a control rod, drive rod and a rod control cluster (RCC) assembly. The total weight of this assembly is available to provide a downward force when it is desired to insert the control rods. All components of the control rod assembly and those components which guide it are referred to as the driveline.

BACKGROUND

Rod insertability is determined by the weight of the control rod assembly and by the various drag forces which act on the assembly. To minimize the mechanical drag, clearances exist between the control rod assembly and its interfacing components. If stresses remain elastic during the seismic event, operability of the driveline after the event is assured because the components return to their initial relative positions. Even if some amount of permanent deformation does occur, operability is still assured unless the resulting misalignment is large enough to generate interference loads which are larger than the weight of the control rods. Operability during the seismic event is assured if the misalignments generated between the driveline components by the event are less than the clearances which are provided. Even if the dynamic displacements are larger than the available clearances and impacting occurs between the various parts of the driveline, operability is still assured if the average drag force is less than the weight of the assembly.

The primary locations where relative displacements potentially affects the misalignment of the driveline are:

- Control rod drive mechanism and rod travel housing-drive rod clearance
- Vessel head penetration - drive rod clearance
- Control rod guide tube - control rod
- Upper and lower core plate lateral displacement
- Fuel assembly - control rod clearance

The fuel assemblies and control rod guide tubes provide guidance and lateral support for the control rods. There is a significant amount of data on RCC's which demonstrates insertability under conditions of static and dynamic lateral misalignments.

CONTROL ROD DRIVELINE DESCRIPTION

- Control Rod Drive Mechanisms

The CRDM pressure housing is a relatively flexible cantilever structure. Basically the CRDM housing consists of the head adapter, latch housing and rod travel housing. The CRDM housing is supported at the bottom of the head adapter by a weld to the inside of the closure head. Threaded on to the top of the head adapter is the latch housing. On top of the latch housing is the rod travel housing which is also connected by a threaded joint and sealed with a canopy weld. There is an upper support on the CRDMs system. Each CRDM is linked to the control rod by a drive shaft which is magnetically driven within the mechanisms. Core shutdown is achieved by releasing the electrical currents and allowing the rod to fall into the core under the influence of gravity.

- Control Rod Guide Tubes

Over most of their length, the guide tubes are large pipes which protect the control rod assembly from the effects of the lateral flow in the upper plenum. Because of this design, mechanical friction due to contact is negligible.

The continuous guidance of the control rods is employed only near the top of the core. This is used to minimize misalignment between the fuel and the control rods so that operability is enhanced. Motion of the guide tubes relative to the fuel assemblies is minimized by locating both components to a common point, the upper core plate; and by minimizing the radial gaps between the reactor internals and the reactor vessel.

SUMMARY OF AVAILABLE TEST DATA

As previously discussed, rod insertion is determined by the weight of the control rod assembly relative to the various drag forces which may be generated between the movable and stationary components of the driveline. Because of the difficulty in analytically determining the magnitude of the friction forces which may exist, tests on actual drivelines are used to provide confirmation of acceptable performance. Primarily, the rod drop tests which are performed prior to plant startup are used to provide this confirmation. Since these tests are performed with and without flow and since each of the active drivelines is tested, these tests provide adequate assurance of acceptable driveline performance under typical operating conditions. For plants similar to Watts Bar units, the small magnitude of these values indicates that the driving force (weight) is much larger than the drag forces normally acting on the driveline.

Laboratory testing has been performed to demonstrate the insertability of RCC type drivelines under various types of static and dynamic misalignments. The results obtained from these various tests support the conclusion that operability can be assured for the Watts Bar units during and subsequent to postulated seismic event. A brief summary of the more significant tests is given in Table I. A discussion of the key results from the various tests follows.

- During the various driveline prototype tests programs, items 1, 2 and 3 of Table I, similar evaluations were conducted for plants having RCC style control rods. As expected, the results further demonstrated the acceptability of driveline performance at various operating conditions.

- For postulated loss of coolant accidents, the control rod guide tubes which are closest to the outlet nozzles in RCC type plants experience very large hydraulic forces. Testing was performed to determine the maximum local lateral deflection for which control rod insertion would occur in plants having 17x17, RCC style drivelines. These tests were performed using 96 inch long and 150 inch long

guide tubes. The data indicated that control rod insertion occurred for lateral deflections of up to 1.0 inch for the 96 inch guide tubes, such as used in the Watts Bar plants. Since the guide tube deflection during the seismic event is expected to be much less than the above mentioned values, seismically induced guide tube deflections would not affect the insertion of the control rods in the Watts Bar units.

- Seismic testing has been performed to demonstrate the performance of the driveline under conditions of dynamic excitation by a Westinghouse licensee. This testing was performed using a single full-size driveline, including a CRDM and rod travel housing, guide tube, fuel assembly, drive rod and an RCC style control rod. During this testing, both sinusoidal and seismic-like accelerations were applied to the driveline. The testing was performed for a range of input acceleration levels and accelerometers were used to measure the response of the various components of the driveline. Rod drop time measurements were made before, during and after testing to confirm the operability of the control rods. Input accelerations were applied at points corresponding to the fuel assembly, vessel supports and the top of the rod travel housing. For the sinusoidal excitation tests, the frequency of the input was varied to match the natural frequencies of the CRDM/rod travel housing, guide tube and fuel assembly. The results confirmed that the control rods could still be easily inserted into the core. This conclusion was further reinforced by the tests which were performed using the seismic wave input and which also demonstrated that the control rods easily fall into the core under the seismic conditions.

TABLE 1

TEST REPORTS RELATED TO DRIVELINE OPERABILITY

	TEST	REFERENCE	VARIABLES CONSIDERED	CONTROL ROD TYPE	
(1)	Indian Point LOPAR Tests	WCAP-7176	Pressure, Temperature, Flow	RCC Control Rods	15 x 15
(2)	17 x 17 Driveline Tests	WCAP-8446	Pressure, Temperature, Flow	RCC Control Rods	17 x 17
(3)	17 x 17A Driveline Tests	WCAP-8788	Pressure, Temperature, Flow with Internal Misalignment and Different Guide Tube.	RCC Control Rods	17 x 17
(4)	KEP Static Deflection Tests (of CRDM)	WCAP-7427	Lateral Deflection of Rod Travel Housing.	RCC Control Rods	14 x 14
(5)	17 x 17 Scram Deflection (Static)	WCAP-9251	Guide Tube Lateral Deflection Fuel Assembly Misalignment.	RCC Control Rods	17 x 17
(6)	CRDM Vibration Testing	MHI Proprietary	Acceleration Level Seismic Input (Sinusoidal and Seismic).	RCC Control Rods	14 x 14
(7)	CRDM Vibration Testing	MHI Proprietary	Acceleration Level Seismic Input(Sinusoidal and Seismic).	RCC Control Rods	17 x 17

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ATTACHMENT III

PVP - Vol. 182

SEISMIC ENGINEERING — 1989
Design, Analysis, Testing, and Qualification Methods

presented at

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SEISMIC TEST AND ANALYSIS FOR PWR REACTOR CORE INTERNALS

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ABSTRACT

The seismic reliability proving test of PWR reactor core internals was carried out.

Fuel assemblies showed complex vibration behavior due to collision in group.

The measured test results of fuel assemblies were compared with the simulation analysis using computer codes.

The analysis showed a good agreement with test results, such as fuel assembly vibration displacement, grid permanent deformation etc.

In S_2 wave vibration test, some local deformation was produced on the grid structure but it was clarified not to be effective to the control rod insertion and the capability of eliminating the decay heat of the core region.

1. NOMENCLATURE

CRDM	: Control Rod Drive Mechanism
F/A	: Fuel Assembly
FINDS	: Computer Code for Fuel Inelastic Deformation under Seismic Condition
FUVIAN 4	: Computer Code for Fuel Vibration Analysis
RCC	: Control Rod Cluster
S_1	: Maximum Design Earthquake
$S_2(1)$	: Extreme Design Earthquake in standardization
$S_2(2)$	: Extreme Design Earthquake in PWR 4Loop Plant Design Wave
$[A]$	: Mode Matrix
$\{D\}$	: Drag Force Vector
$[C_g]$, $[C_r]$	: Grid and Rod Damping Matrix
E	: Grid Impact Energy
Esc	: Energy Storage Capacity (Thresheld Energy of Grid Deformation Progress)
$\{F_r\}$	: Impact Force Vector
$[H]$	: Damping Factor Matrix
$\{H_y\}$	: Hysterisis Force Vector
K_g, K_r	: Grid and Rod Stiffness
$[M]$, $[C]$, $[K]$	: Mass, Damping and Stiffness Matrices
$\{P\} = -[M]\{\ddot{X}_o\}$	
$\{U\}$	: Mode Intensity Vector
$\{X\}$	: Nodal Displacement of Beam
$\{\ddot{X}_o\}$	: Seismic Acceleration of Barrel
Z	: Displacement of Grid
δ^o	: Initial Gap between Adjacent F/As
δ^p	: Gap Increase due to plastic Deformation
$[\Omega]$	: Natural Angular Frequency Matrix

2. INTRODUTION

Since Japan is a country that has frequent earthquakes, integrity in earthquake is regorously required in the construction of nuclear power plants.

Consequently, the soundness of the structure and the component must be ensured based on strict anti-seismic requirements that in ordinary cases, include anti-seismic design, test and inspection before plant operation.

On the other hand, the public attitude toward reliability on nuclear power plant operation is not very stable at the present stage.

For the purpose of proving seismic reliability, Nuclear Power Engineering Test Center (NUPEC) has curried out the seismic reliability proving tests of nuclear power plant facilities, sponsered by the Ministry of International Trade and Industry (MITI).

The seismic proving tests intend to prove ability to maintain seismic reliability and the integrity of the nuclear power plant with a sufficient margin in the occurence of the large earthquakes, using the large scale high perfomance vibration table of Tadotsu Engineering Laboratory.

PWR reactor core internals proving test was conducted third among them.

The purpose of the test is to prove seismic safety and reliability of the reactor core internals and also to confirm adequacy of the present seismic analysis method being used in the current seismic design for the PWR reactor core internals.

~~Confirmation of insertion function of control rods during the time of earthquakes is of prime importance since the reactor core system is expected to exhibit complex vibration behavior due to collision of fuel assemblies.~~

This paper focused on the dynamic behavior of F/As under design-earthquake conditions.

3. TEST

Test Facility

In order to prove the seismic reliability and integrity of the nuclear power plant, a full scale or close to the actual size of components is preferable.

The large-scale high-perfomance vibration table of NUPEC was designed in Tadotsu to meet this purpose. This is the largest scale vibration table in the world.

b. Seismic Wave

The basic design earthquake ground motion waves S_1 and S_2, which had been improved and standardized by MITI for the high seismic zones and a plant design wave, were used as input to the reactor building analysis model for standard PWR plants to obtain the floor response waves at the level of the reactor vessel support.

Then among these response waves, the ones giving the severest effects on the reactor core internals were adopted as the input waves of the provig test.

The conditions of the input waves are shown in Table 1.

c. Test Model

The test component is a full scale model of the reactor core portion of the reactor core internals for 1,100 MWe class improved and standardized PWR plant.

The full scale model was adopted for the reason that the vibration behavior of the reactor core region could be observed and control rod insertion function could be comfirmed in the similar condition to the actual component.

The major components of the test model are 45 F/As, 2 [illegible], 2 guide tubes, 2 CRDMs, core support structure [illegible]uding a lower core support structure and an upper [illegible]e support structure, vessels and an upper structure.

The structural configuration of the test model is shown in Fig. 1.

d. Outline of the Test

(1) Test Conditions

The tests were conducted under the atmospheric pressure and filled with static water.

Excitation directions were both vertical and horizontal simultaneously.

The control rods were inserted during the excitation.

The control rod insertion starting time was determined reflecting the time which covers the generation of the maximum response acceleration of input seismic wave in the duration of control rod insertion to give a severe condition in the evaluation of inserting performance.

(2) Proving Test

The test model was excited with seismic waves of the severest condition expected in the present seismic design, and the control rod insertion test was conducted.

(3) Design Method Confirmation Test

The seismic analysis method used in the design of actual systems and adequacy of the proving test model as a model for the proving test was evaluated.

[illegible]arginal Vibration Test

[illegible]e test model was excited with seismic waves for the proving test with an increased acceleration level to study the seismic safety margin.

Since the $S_2(1)$ wave gives the severest effects on the F/As, the test model was excited with $S_2(1)$ wave, and then it was excited with increased level of the $S_2(2)$ wave up to 1.5 times to confirm the inserting performance of the control rods.

This paper mainly describes the proving test and the marginal test. And the marginal test was an extension to the proving test.

e. Measurement

There are three types of test model configuration as shown in Fig.1: the proving model, the row model and the square model, and for the proving model, locations of the vibration detectors are shown in Fig.2.

The following vibration detectors are installed in the test model, strain gauge, accelerometer, force gauge displacement measuring sensor and fiberscope.

4. RESULTS

a. Proving Test

(1) Results of S_1 and $S_2(2)$ Wave Vibration

Maximum response acceleration and maximum response strain were measured for the reactor core internals. And maximum response values of displacements, acceleration, strain and grid impact force of F/As were measured.

The test model did not show any abnormal response and the stress level was within the range of elasticity for both S_1 wave and $S_2(2)$ wave vibration.

The RCC was reliably inserted with a very small delay time, 0.04 sec for S_1 and 0.02 sec for $S_2(2)$.

(2) Results of $S_2(1)$ Wave Vibration

Since local deformations on the grids of the F/As by visual inspection after the seismic proving test were found, residual grid deformations for each vibration were examined using the grid displacement measurement data. Slight deformation was caused on the 90° side by the first vibration with $S_2(1)$ wave, and on the other hand, slight deformation on the 270° side was caused by the second vibration with $S_2(1)$ wave, but the deformation hardly progressed after the third and subsequent vibration.

The relation between input wave level and impact force is shown in Fig.3.

The results of the inspection after $S_2(1)$, wave level and impact force are that the maximum defromation of approximately 2mm of the grid was observed in seven F/As and the impact force for generating deformation was approximately 3 ton.

Control Rod Inserting Performance

The control rods were satisfactorily inserted, and the inserting time delay was insignificant. In the proving test, the grid of the F/As, subjected to inserting a RCC B, had been locally deformed. It was confirmed that grid deformation to this extent hardly affected on the inserting performance.

Strength of Fuel Assemblies

As for the stress of the F/As in $S_2(1)$ vibration, the maximum value in the control rod guide thimble was 8.4 kg/mm^2, and that in the fuel clad tube was 3.2 kg/mm^2, which are much lower than the yield strengths of 24.6 kg/mm^2 or 62.3 kg/mm^2 respectively.

b. Marginal vibration test

In the repeated excitation with $S_2(1)$ wave, growth of the grid deformation due to repeated input seismic waves which cause the deformation in the grids of F/As, and the effect of deformation on the insertion performance of the control rods was investigated.

Subsequently, excitations were conducted with waves amplified to 1.5 times as large as $S_2(1)$ wave input

level to confirm that the insertion performance of the control rod could be reliably maintained, and to investigate the growth of grid deformation in the amplified input wave vibration.

(1) Response of Fuel Assemblies

Local deformation was caused in the grids of seven at the level of $S_2(1)$ and two additional F/As up to 1.5 times of $S_2(1)$ as shown in Fig.4.

The relationship of the input level and the maximum response displacement in the central part of the F/As (the fifth grid level) is shown in Fig.5.

Displacement of the F/As at both ends increases almost in proportion to the input level, and this tendency remains unchanged before and after generating the local deformations.

Stresses of the F/As were still within the range of elasticity even in the marginal excitation with amplification of 1.5 times as large as $S_2(1)$ wave input level.

(2) Insertion Performance of Control Rods

Fig.6 and 7 show the results of the control rod insertion test in the marginal excitation and the results of the control rod insertion test with various seismic wave excitations.

These figures show that the control rods were inserted within a specified limit time (2.2 seconds) with a sufficient safety margin even in the marginal excitation of 1.5 times amplification.

5. SIMULATION ANALYSIS

a. S_1 Wave Excitation

(1) Analysis Code

Simulation analysis of S_1 wave excitation was conducted with a F/A vibration analysis code FUVIAN 4, which analyzes the time historical vibration response of F/As dealing with the collision between F/As and which introduces the amplitude dependence of resonance frequrecy and damping ratio of F/A.

Analytical Model of FUVIAN 4 is shown in Fig.8.

Response accelerstion waves obtained in the test at the upper and lower core plate levels were directly used as input waves for simulation analysis.

(2) Comparision with Test Results

Comparison between the results of simulation analysis and the measured values, displacement response wave in the forms of time history and the maximum impact force in the proving model in S_1 wave vibration are shown in Fig.9 and 10. From the comparision of the time historical waves, the results of simulation analysis and the measured values show a fairly good agreement, and the impact forces obtained by the analysis are evaluated on the safety side.

Fig.11 shows the results of simulation analysis for the proving model and the design confirmation model with S_1 wave vibration, varying the excitation level. In this figure, it is recognized that the tendency of increasing the impact force as increasing the vibration level in the proving model differs from that in the square model, showing a similar tendecy in the results of measurement.

Fig.12. shows the correlation between the measurement values and the analysis values of displacement and impact force. The analysis values and measured values of impact force show a fairly good agreement, giving slightly larger values in the response analysis. As for the displacement, the analysis and measured values show also a fairly good agreement, giving slightly larger values.

b. $S_2(1)$ Wave Excitation

(1) Analysis Code

FINDS code was used for the simulation analysis of $S_2(1)$ wave excitation and the repeating excitations including excitation with 1.5 times amplified $S_2(1)$ wave.

The process to calculate the vibration resporse in FINDS is the same as FUVIAN 4.

FINDS code has the additional function to calculate the grid plastic deformation when the grid impact energy exceeds the elastic limit considering the change of grid stiffness and the change of gaps between two adjacent F/As due to grid plastic deformation.

Fig.13. shows the basic flow diagram of FINDS code.

(2) Comparison With Test Result

$S_2(1)$ wave excitation simulation analysis

Simulation analysis was conducted with the response acceleration waves at the upper and lower core plate levels obtained in the test, and the analysis results were compared with the test results.

Fig.14. shows the grid impact force and the initiation and the progress of grid deformation of the center (5th) grids of 90° side end F/A and 270° side end F/A in the form of time history.

On the initiation point and the process of the progress of grid deformation, the analysis results and the measured results show a good agreement.

The impact force time history by the analysis also shows a good agreement with the test data.

Simulation analysis of the marginal vibration test was also performed to investigate of the simulatability of the analysis.

The accumulated grid deformation after each excitation and the magnitude of grid deformation in each excitation are compared in Fig.15.

Simulation analysis shows a good agreement with the test results on the progress of grid deformation and produced impact forces and displacements of F/As, and concerning the maximum values of these response results, simulation analysis gives slightly larger value than the test results.

c. Simulatability of the codes

Simulation analysis on S_1 wave excitation and $S_2(1)$ wave excitations by FUVIAN 4 and FINDS show the good simulatability of the analysis not only on the F/A vibration response such as displacement and impact force but also on the grid deformation which might be produced when the input wave is severe, and also show the adequate conservatism on the maximum response value which gives the usefulness to the codes for design use.

6. CONCLUSIONS

Seismic proving test of PWR reactor core internals using the large-scale and high-performance vibration table was conducted in 1985, and the seismic strength and RCC inserting function using the full scale test model of the reactor core internals was verified.

Subsequently, the detailed analysis and evaluation of the test data obtained by the test was carried out in the latter half of 1985, and the seismic safety of the actual PWR reactor core internals was evaluated.

With 1.2 S_2 and 1.5 S_2 wave inputs under the conditions which exceed the design conditions, RCC insertion function was verified to be maintained.

In S_2 wave vibration test, impact force exceeding the elastic limit of the fuel grid was generated, and consequently, some local deformation was produced on the grid structure but it was clarified not to be

ective to the RCC insertion.

imulation analysis by FUVIAN 4 and FINDS showed a d agreement with test results, thus the validity of Codes for fuel seismic analysis was confirmed.

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1. Omori, T et al, "Proving Test on The Reliability for Nuclear Power Plant, PWR Reactor Core Internals" NUPEC, March, 1987.
2. Sato, T et al, "The FINDS CODE for Fuel Seismic Analysis Considering Inelastic Impact Behavior" Transactions of the ASME 1988 PVP Conference.

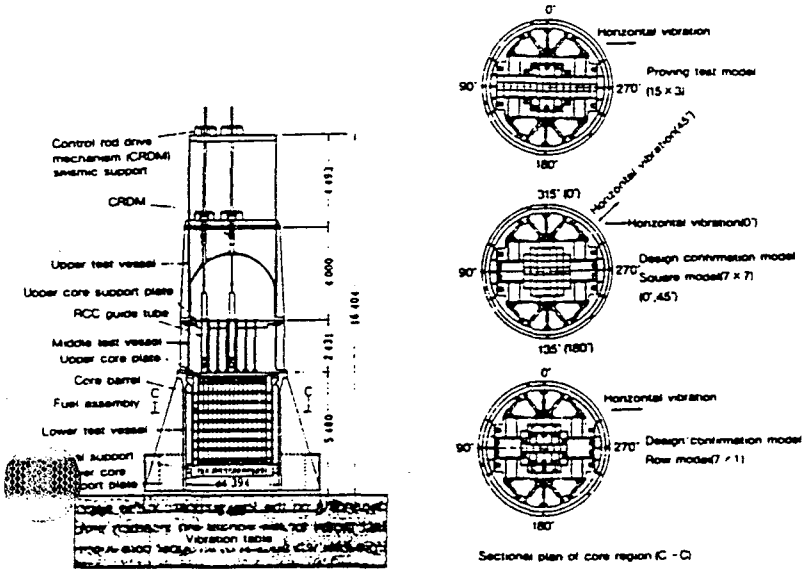

Fig. 1 Configuration of PWR Reactor Core Internals

Table. 1 Seismic Wave for Proving Test (PWR reactor core internals)

Seismic wave	Direction	Duration (sec)	Max. acceleration (Gal)
S_1 Floor response*	Horizontal, Vertical two directions	20	Horizontal : 408.0 Vertical : 150.6
S_1 Floor response(1)**	"	40	Horizontal : 729.0 Vertical : 251.9**
S_1 Floor response(2)**	"	40	Horizontal : 714.1 Vertical : 375.3

* S_1 . $M = 8.4$, $\Delta = 90$ km.
Phase characteristics TAFT EW (1952).
$V_s = 1\,000$ m/s
S_1 : (1) $M = 8.5$, $\Delta = 68$ km.
Phase characteristics Hachinoe EW(1968).
$V_s = 1\,000$ m/s
(2) PWR 4 loop plant design wave

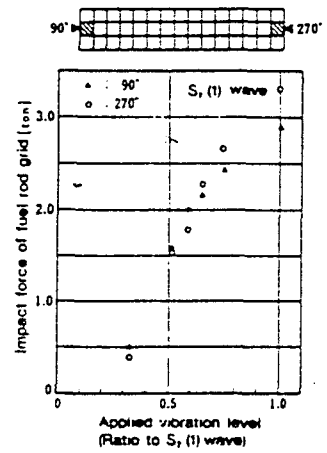

Fig. 3 Relations between Applied Vibration Level and Impact Force of Fuel Grid

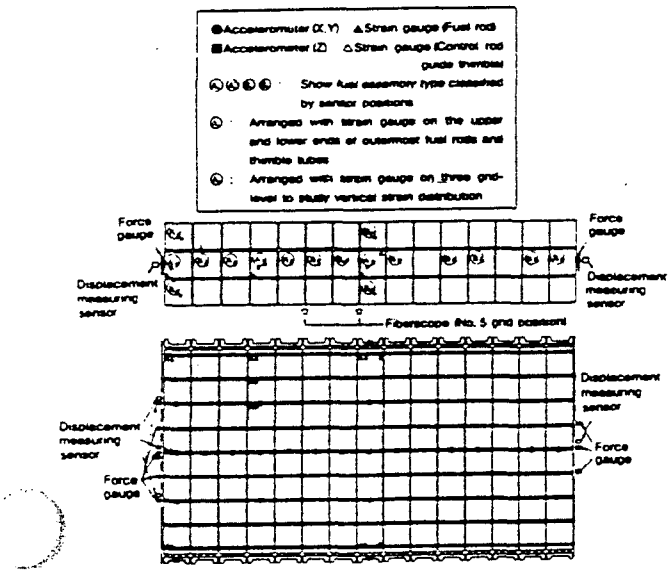

Fig. 2 Measuring Positions of Vibration Detectors on Proving Test Model (core region)

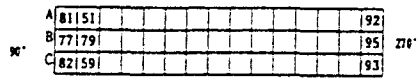

Fig. 4 Fuel Assembly Positions of Grid Deformation Occurrence in Marginal Vibration Test

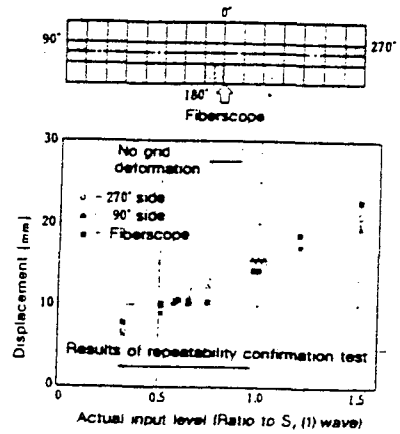

Fig. 5 Relation between Input Level and Response Displacement of Fuel Assembly in Marginal Vibration Test

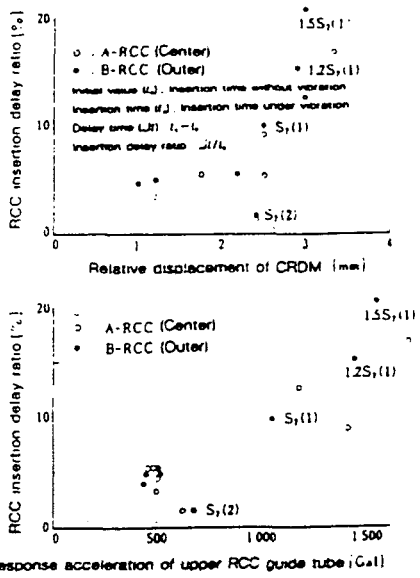

Fig. 6 Relation between Test Model Response and RCC Insertion Delay Ratio (No. 1)

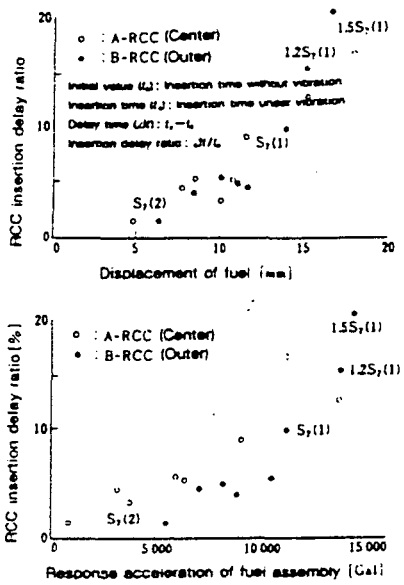

Fig. 7 Relation between Test Model Response and RCC Insertion Delay Ratio (No. 2)

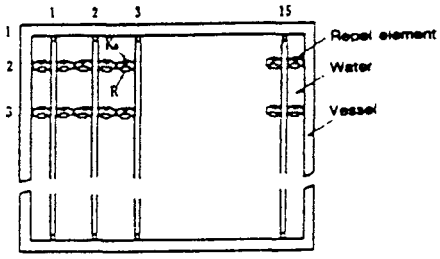

Fig. 8 Vibration Analysis Model

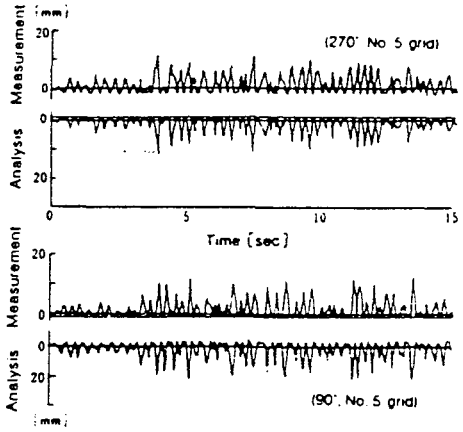

Fig. 9 Comparison between Measurement and Simulated Analysis by FUVIAN4 (15x3 core, S_1 wave vibration, time histrical displacement)

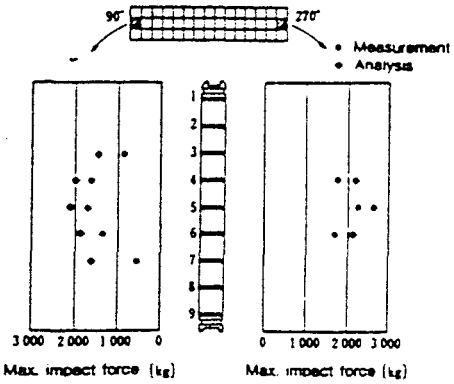

Fig. 10 Comparison between Measurement and Simulated Analysis by FUVIAN4 (15x3 core, S_1 wave, Max. Impact force)

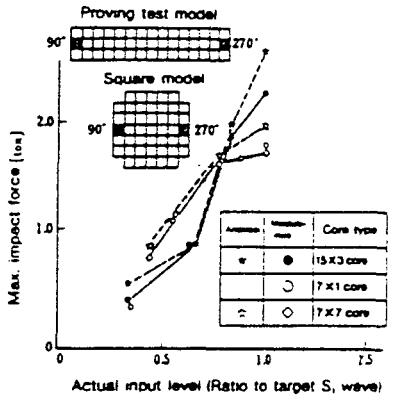

Fig. 11 Comparison of S_1 Wave Seismic Response between Proving Model and Square Model (Relation between input level and impact force of fuel grid)

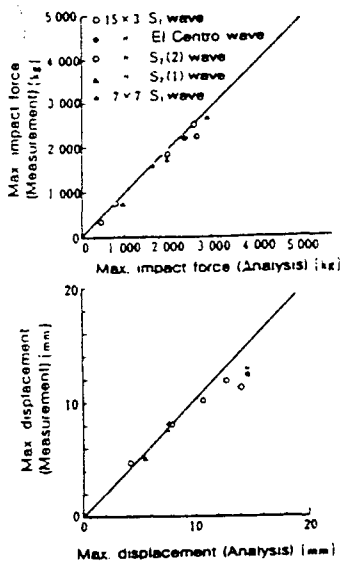

Fig. 12 Comparison between Measurement and Analysis by FUVIAN4

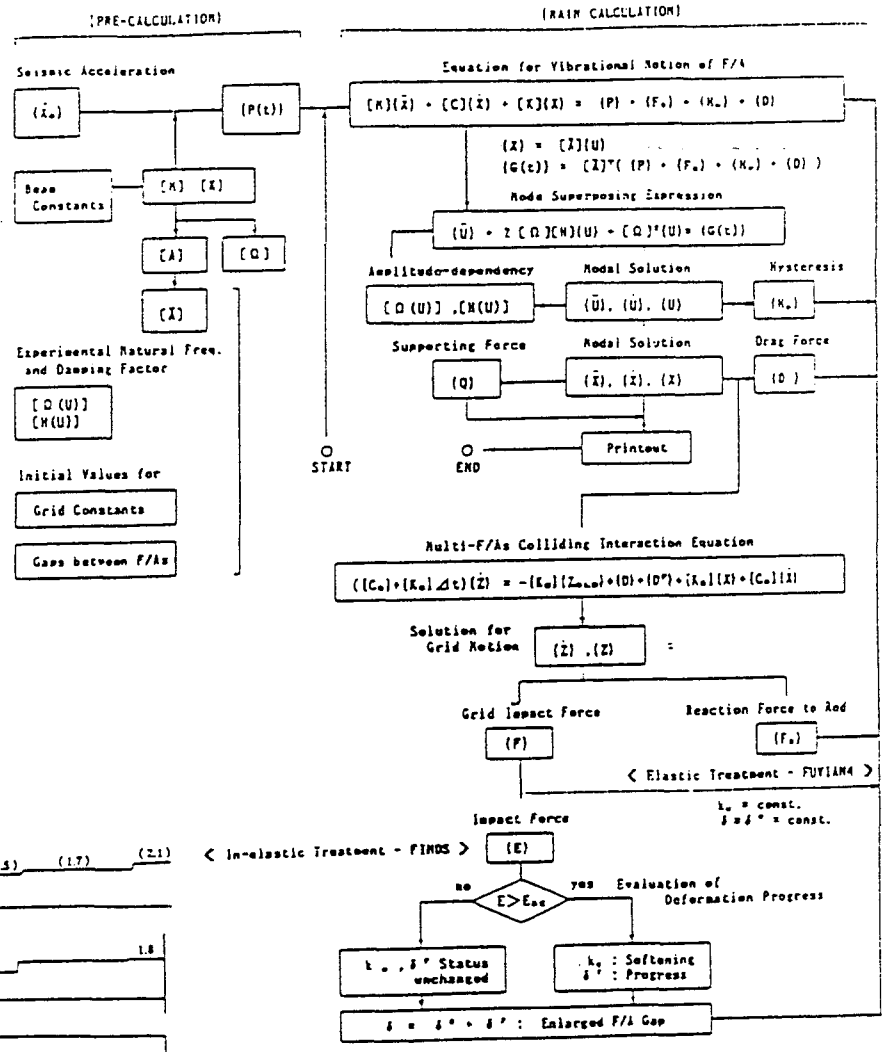

Fig. 13 Basic Flow Diagram of FINDS Code

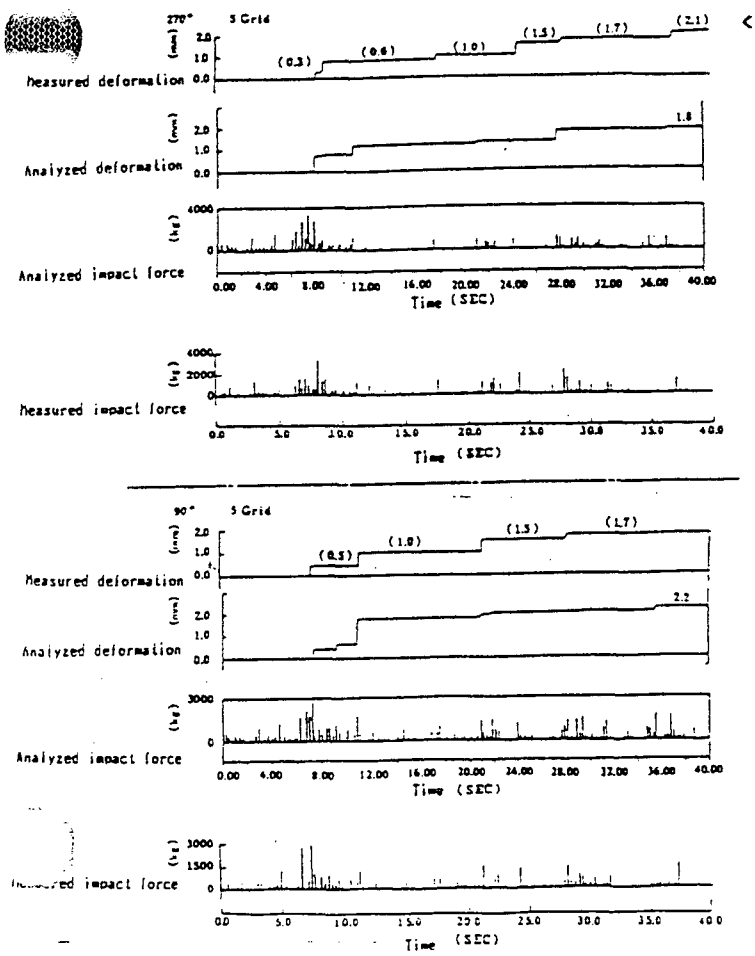

Fig. 14 Time history of grid impact force and progress of grid deformation

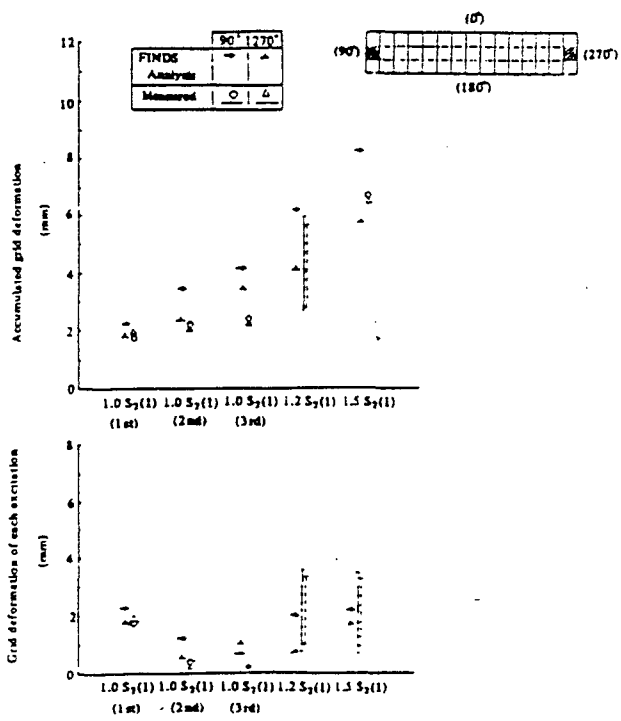

Fig. 15 Grid deformation change of 5 grid of end fuel assemblies in marginal excitation

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Attachment 4

number of challenges to the protection system and the number of transients to which the plant is subjected has important benefits in terms of both safety and plant availability.

The NRC has reviewed and approved the elimination of this trip function based on the use of the Median Signal Selector in the ADFCS.

AMSAC

Prairie Island has chosen a logic design for its AMSAC system that uses feedwater flow and turbine load as inputs. However, these inputs are already available in the ADFCS. It has thus proved to be expedient and cost-effective to implement the AMSAC function within the ADFCS Distributed Processing Family (DPF) system.

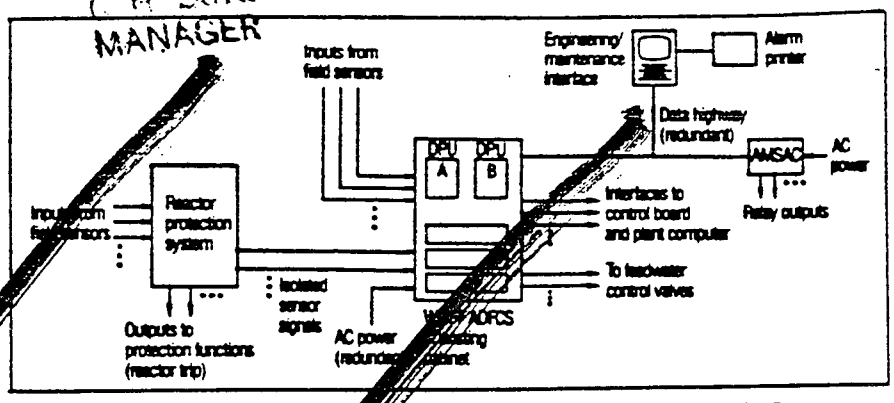
AMSAC will be located in a separate existing cabinet which will contain a small programmable logic controller that is part of the DPF line that interfaces to the data highway system. Redundant data highway cables will be run from the feedwater cabinet to the AMSAC. The variables needed by AMSAC will be passed over the data highway and the AMSAC logic will be executed in the programmable controller. The cabinet will also contain a test panel and interposing relays for the required actuation outputs. The NRC has approved this AMSAC design.

PUTTING IT TO THE TEST

A comprehensive and integrated analysis and testing process has been devised for this system, starting with the plant modelling and setpoint study and ending with plant startup testing.

In the first stage of the process, a plant specific simulation model was developed that included the new feedwater control algorithms. This model was used to perform transient analysis studies to select control system setpoints and verify system performance.

The results of this activity were also used in the next phase of testing which



▲ **Prairie Island's digital feedwater control system upgrade with AMSAC. DPU -distributed processing units. DPF - Westinghouse Distributed Processing Family. ADFCS -Advanced Digital Feedwater Control System.**

was hardware-in-the-loop testing. In this phase the control algorithms were programmed into the DPF equipment which was interfaced to the plant simulation model through analog signal interfaces. The purpose was to verify that the control algorithms, as implemented in the DPF equipment, provide the same response as was obtained in the setpoint study. The results of the setpoint study were used as acceptance criteria for this hardware-in-the-loop testing.

PROGRESS SO FAR

Extensive factory acceptance testing of the system was performed prior to shipment, with further pre-operational testing being carried out once the system had been installed in the plant. Finally, operational tests were conducted during the plant startup and power operations.

All plant modifications for this upgrade were accomplished within a normal refuelling/maintenance outage. These were carried out for unit 2 in April 1989 and for unit 1 in February of this year.

All routine periodic surveillance testing of the reactor protection system channels from which some of the inputs to the ADFCS are derived, is now carried out with the ADFCS in AUTO mode. This has been made possible by virtue of the signal selector algorithms, and has made testing more efficient with less risk of a disturbance to the plant.

The ADFCS has provided effective automatic control throughout the range of the power operation. This has proven to be particularly important for unit 2 which experienced several forced outages during the first year of ADFCS operation. The ADFCS continued to provide effective control performance during each plant startup, thereby alleviating the operator burden and contributing to efficient plant operation and return to power.

In addition, two other plant operating experiences during the first year of ADFCS operation have also demonstrated the reliability of the new system:

- A heater drain pump tripped on Unit 2 while at full power. This caused a feedwater system disturbance and a runback in plant load. The ADFCS responded well to this event, keeping the steam generator levels under good control.
- An ADFCS analog input card, which contained five analog input signals, had to be replaced. This was done with the system on-line in AUTO mode without any disturbance to plant operations. The signal selector algorithms prevented any of these failed channels from affecting the control system.

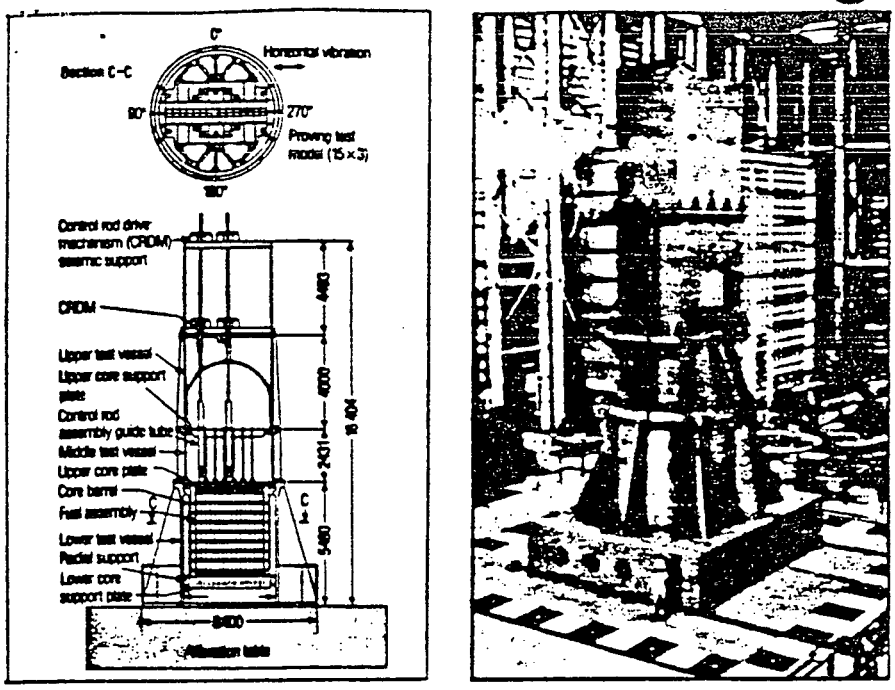
Control rod behaviour in earthquakes

By S. Kawakami, H. Akiyama, H. Shibata, M. Watabe, T. Ichikawa and K. Fujita

For some years the Japanese have been working on a major research programme to determine the likely effects of an earthquake on nuclear plant internals. One aspect of this was a study of the behaviour of PWR control rods as they are being inserted in the core.

These days almost every nuclear facility is designed to withstand earthquakes. In many countries this is considered simply as a sensible precaution and few people expect that the plant will ever experience a major earthquake. In Japan, on the other hand, earthquakes are a common occurrence and cause structural damage or loss of life almost every year.

The Great Kanto earthquake of 1923, which had a magnitude of 7.9, killed 140 000 people and destroyed many of



▲ Fig 1. PWR core internals model on the Tadotsu shake table.

the buildings in the cities of Tokyo and Yokohama. Historically, earthquakes in this region have a return period of about 70 years and so there is a high probability of another major earthquake along the Pacific Coast in the next few years.

Several nuclear power plants are situated along this coast and it is therefore reasonable to expect that some of them will be exposed to a severe earthquake during their operating lifetime. The situation is much the same in other parts of the country. Not surprisingly, therefore, Japan takes seismic design very seriously and spends quite large sums on seismic research and testing.

This article is concerned with the effect of the earthquake motion on the control rods as they are being inserted into the core of a PWR. This is just one aspect of a series of tests carried out at the Tadotsu Engineering Laboratory to find out how PWR reactor internals would behave under severe earthquake conditions.

ROD INSERTION QUESTION

In a PWR, the control rods are clusters of absorber rods which enter into guide thimbles in the fuel assemblies. Typically the clearance between the rod and the guide thimble is about 0.88mm.

In the core, the fuel assemblies are held in position at the top and bottom. However the assemblies have some flexibility and can bend sideways to some extent. In a small earthquake the assemblies will vibrate from side to side independently. At higher levels of excitation, impact will occur between grids of adjacent assemblies. During a strong earthquake, the assemblies will vibrate in a more or less synchronized fashion, with frequent impacts both between the grids and between the grids of the peripheral assemblies and the core baffle. In the worst condition, the assemblies would be deflected by an amount equal to the sum of the gaps between grids. At the widest part of the core this total is about 15mm.

During an earthquake, therefore, the control rods have to enter guide thimbles which are being deflected from side to side and the control rods themselves must bend to conform to these deflections. The control rods are quite flexible and there is no doubt that they will enter the core. The question is to what extent the control rod drop time will be affected.

WORK AT TADOTSU

The Tadotsu Engineering Laboratory has a very large vibration table belonging to NUPEC (Nuclear Power Engineering Test Center). NUPEC is an extra-governmental organization and the verification tests carried out at the laboratory are sponsored by MITI.

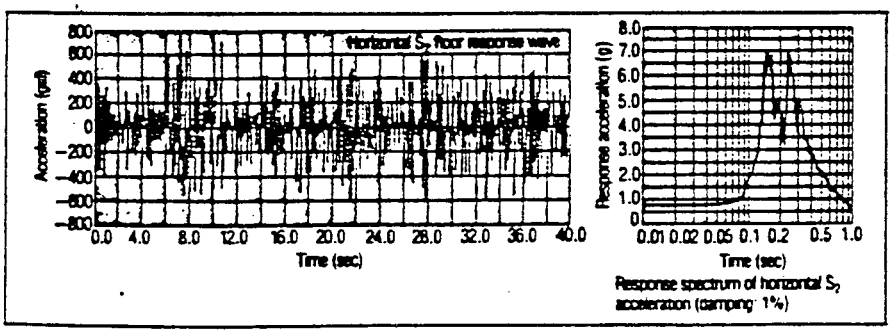
The vibration table is 15m × 15m and has a maximum loading capacity of 1000t. Horizontal vibrations are caused by seven hydraulic actuators, each with a capacity of 450t and a stroke of ±200mm. There are 12 vertical actuators, each with a capacity of 300t and a stroke of ±100mm. The actuators are computer controlled to produce seismic waveforms.

Since 1982, the laboratory has been engaged in a programme of testing large scale models of LWR components such as containment vessels, pressure vessels, core internals and so on.

PWR core internals model. This model was a full scale model of the core internals of a 1100MWe PWR. It included the upper and lower internals packages and the upper part of the

Seismic input waves

Seismic wave	Direction	Duration (secs)	Max. acceleration (gal)
S_1 floor response	Horizontal and vertical	20	Horizontal 408.0 Vertical 150.6
S_2 (1) floor response	Horizontal and vertical	40	Horizontal 729.0 Vertical 251.9
S_2 (2) floor response	Horizontal and vertical	40	Horizontal 714.1 Vertical 375.3



▲ Fig 2. Example of an S_2 horizontal wave.

S. Kawakami is with the Nuclear Power Engineering Test Center. H. Akiyama and H. Shibata are with the University of Tokyo. M. Watabe is with Tokyo Metropolitan University. T. Ichikawa is with Mitsubishi Atomic Power Industries (2-4-1 Shibaura, Minato-ku, Tokyo 105, Japan) and K. Fujita is with Mitsubishi Heavy Industries.

reactor vessel. Two sets of operational control rod drive mechanisms were included, together with associated guide tubes and control rod clusters.

Forty-five fuel assemblies were provided and these could be arranged in different configurations. The control rod drop tests were done with a configuration of three rows of 15 assemblies, which is the maximum number of assemblies in any row of a 1100MWe core. This configuration allows the largest vibrational displacement of the fuel assemblies and consequently, the most severe condition for control rod insertion.

The model was filled with water during the tests but was not pressurized.

Some idea of the size of the model can be seen in Fig 1 opposite. The total weight was 550t and the height was over 50ft.

Seismic waves. Japanese nuclear power plants are designed to withstand two earthquake ground motion waves S_1 and S_2. The S_1 earthquake is the "maximum design earthquake" and is the largest earthquake which can be expected in the light of historical records and similar considerations.

The S_2 earthquake is the "extreme design earthquake" and is the maximum conceivable earthquake for the region based on tectonic structures and an earthquake (M=6.5) directly beneath the site.

For the tests, the standard S_1 and S_2 ground motions for high seismic zones were used as inputs to the reactor building analysis model of a standard 1100MWe PWR to obtain the floor response waves at the level of the reactor vessel supports. These wave forms were used for the tests and their main features are shown in the table, while Fig 2 shows an example of an S_2 horizontal wave (gal is a unit of acceleration used in seismic studies and is equal to 1cm/sec^2).

There are actually two versions of the S_2 wave, one gives the most severe conditions for the fuel assemblies and the other gives the most severe effect on the control rod drive mechanisms.

Vibration tests were carried out using the S_1, $S_2(1)$ and $S_2(2)$ waveforms, and also with $1.2 \times S_2(1)$ and $1.5 \times S_2(1)$ waveforms. The control rods were released into the core at the time when response accelerations were at their maximum values.

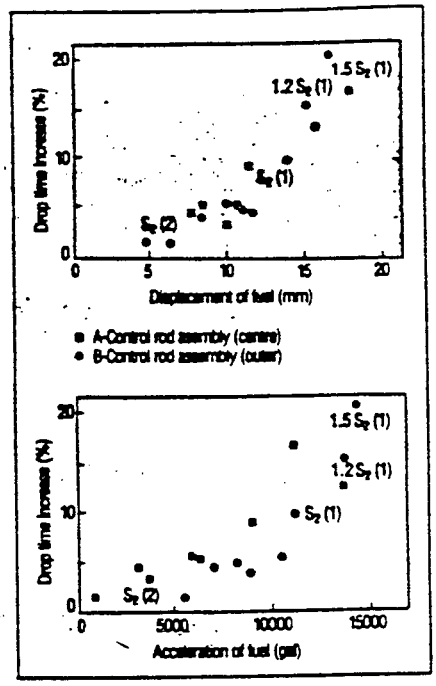
TEST RESULTS

Control rod drop time test results are shown in Fig 3. The data are plotted as the percentage increase in drop time relative to the normal drop time.

As can be seen, the seismic forces have a relatively small effect on control rod drop time and even for the most severe earthquake, the drop time is within the specified time used in safety analyses.

Other test results. The tests also showed that the seismically induced stresses in the core internals and control rod drive mechanisms are relatively small and adequate seismic safety margins already exist in the present designs.

The only stresses which exceeded the elastic limit were in the grids of seven fuel assemblies and the largest grid deformation was about 2mm. These deformations occurred during the $S_2(1)$ test. The fuel assembly containing the B control rod assembly was one of these seven fuel assemblies and no effect on the rod drop time was observed as a result of the grid deformation.



▲ **Fig 3. Control rod drop time test results. The acceleration of the fuel means the maximum acceleration of the fuel assembly measured in cm/sec^2. In practice this occurred at the midpoint of the fuel assembly length. It is much higher than the ground acceleration due to flexibilities in the building, reactor structures and fuel assemblies themselves.**

Upgrading Beznau with a distributed information system

Westinghouse is installing a state-of-the-art, distributed I&C system at Beznau in Switzerland, in a three-phase programme that is due to be completed in the autumn of 1994. The system will collect and process data from 1400 analog and 3200 digital signals throughout the plant.

Westinghouse Electric is supplying a distributed microprocessor based plant information and control system to the Beznau plant of Nordostschweizerische Kraftwerke (NOK) in Switzerland. The Anlage Information System (ANIS) consists of a series of stations, or "drops" located around the plant and will process 1400 analog and over 3200 digital signals, with the ability to increase the addressable point count to 32 000. The system includes:

- Data acquisition functions.
- Computation server functions.
- Operator station functions.
- Data storage retrieval functions.
- Logger functions.
- Data link functions.

The goal of this system is to integrate the extensive information available in the plant into a centralized database. Each of the individual drops is connected to the rest of the system by a dual highway architecture. Database Highway is used for acquisition of plant sensor data and for passing computed values. The second highway is used for passing computed values and messages between the calculational stations and the operator stations, which are available throughout the main and emergency