Reactor, Units 2 and 3, Fukushima Daiichi NPS

Existence of the impact by aged deterioration immediately after the occurrence of Tohoku-Chihou Taiheiyou-Oki Earthquake

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Tokyo Electric Power Company

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(1) Item by item verification of aged deterioration

(2) PLM earthquake-proof safety evaluation from the earthquake response analysis result of major facilities by the observation record of Tohoku-Chihou Taiheiyou-Oki Earthquake

Item by item verification of aged deterioration

In addition to the daily maintenance (maintenance program), extracted the part with least margin by the anti-earthquake safety evaluation result taking account of the aged for deterioration items (6 items)* and checked the existence of impact by the seismic motion of Tohoku-Chihou Taiheiyou-Oki Earthquake (“ Earthquake ”) * Please refer to the guideline for implementation of the countermeasures for aging management (NISA, April 16, 2010)

Aged deterioration	Unit	Part for earthquake-proof safety evaluation	Subject part	Require earthquake-proof safety evaluation from the seismic motion of Earthquake?
Low cycle fatigue	Unit 2	Reactor Pressure Vessel	Feedwater nozzle	○
	Unit 3	Primary Containment Vessel	Bellows, feedwater line piercing part	○
Radiation-induced stress corrosion cracking	Units 2 & 3	Structures in the Reactor	Upper grid plate	○
Neutron irradiation embrittlement	Units 2 & 3	Reactor Pressure Vessel	Body	○
Thermal aging of 2 phase stainless steel	Units 2 & 3	Primary Loop Recirculation System Pump	Casing	×
Insulation degradation of electrical facility and instrumentation	Units 2 & 3	LV cable etc	–	×
Deterioration of strength and shielding of concrete	Units 2 & 3	Reactor Building etc	–	×

Impact by the Earthquake is significant because of the vibration response characteristics / structure and strength

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The rationale in determining there was no impact by the aged deterioration immediately after the Earthquake

Aged deterioration item	Part for anti-earthquake safety evaluation	Rationale
Thermal aging of 2 phase stainless steel	Primary Loop Recirculation System Pump	・As for the thermal aging on the casing of Primary Loop Recirculation System Pump, from the embrittlement test results at domestic and overseas and inspection, there was no possibility of actualization at the time of the Earthquake. Therefore, we determine that there was no impact by the Earthquake.
Insulation degradation of electrical facility and instrumentation	LV cable etc	・Insulation degradation assumed on LV cables etc does not have impact on the anti earthquake resistance such as the mass of equipments. Therefore, we determine that there was no impact by the Earthquake.
Deterioration of strength and shielding of concrete	Reactor Building etc	・As for the anti-earthquake evaluation of concrete at Reactor Building etc, we confirmed soundness at the design condition (without aged degradation) ・On the other hand, we confirmed that the assumed aged degradation on concrete does not have impact on the strength degradation of concrete by analysis and tests. ・Therefore, we determine that the soundness of concrete with the aged degradation is confirmed from the evaluation result at the design condition.

(1) Fatigue crack

Fatigue evaluation result of Reactor Pressure Vessel (feedwater nozzle), Unit 2

(PLM evaluation figure)

Part for evaluation	Fatigue accumulated coefficient based on actual operation	Fatigue accumulated coefficient by seismic motion (S_2 seismic motion)	Total (Allowance < 1)
Feedwater nozzle	0. 434	0.010	0.444

Fatigue evaluation result of Reactor Pressure Vessel (bellows, feedwater line piercing part), Unit 3

(PLM evaluation figure)

Part for evaluation	Fatigue accumulated coefficient based on actual operation	Fatigue accumulated coefficient by seismic motion (S_2 seismic motion)	Total (Allowance < 1)
bellows, feedwater line piercing part	0. 611	0.020	0.631

(1) Fatigue crack

- Change of seismic motion affects the calculated cumulative fatigue factor by the seismic motion only.
- The evaluated figure in the PLM evaluation document (cumulative fatigue factor by S2 seismic motion) is small enough. We determined that the figure will not exceed the allowance even taking account of the seismic motion by the Earthquake.

We determined that immediately after the Earthquake, there was no impact on the soundness of the facility from the viewpoint of fatigue crack.

(2) Radiation-induced stress corrosion cracking of the upper grid plate

The assumed cumulative dose to the upper grid plate, Reactor, Unit 2

(n/m^2)

Part for evaluation	The assumed cumulative dose at the time of the Earthquake
Upper grid plate (SUS316L)	**Approx 2.6×10^{25}**

※ The cumulative dose (threshold dose) for SUS316 on the radiation-induces stress corrosion cracking: $1\times10^{25} n/m^2$

※ We replaced the upper grid plate at the 17th maintenance outage (August 1998-August 1999).

Radiation-induced stress corrosion cracking evaluation result of the upper grid plate, Reactor, Unit 2 (PLM evaluation figure)

($MPa\sqrt{m}$)

Part for evaluation	Expansion coefficient for estimated missing stress	Fracture toughness ※2
Upper grid plate	14.3 ※1	43.2

※1 Assessment 60 years later after the commencement of operation

※2 Lower limit (BWR) for fracture toughness of radiation stainless steel from "Report on technological development for long-life plant" published by Japan Power Engineering and Inspection Corporation.

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(2) Radiation-induced stress corrosion cracking of the upper grid plate

The assumed cumulative dose to the upper grid plate, Reactor, Unit 3

(n/m²)

Part for evaluation	The assumed cumulative dose at the time of the Earthquake
Upper grid plate (SUS316L)	**Approx 2.9×10^{25}**

※ The cumulative dose (threshold dose) for SUS316 on the radiation-induces stress corrosion cracking: 1×10^{25}n/m^2
※ We replaced the upper grid plate at the 16th maintenance outage (May 1997-September 1998).

Radiation-induced stress corrosion cracking evaluation result of the upper grid plate, Reactor, Unit 3 (PLM evaluation figure)

(MPa√m)

Part for evaluation	Expansion coefficient for estimated missing stress	Fracture toughness ※2
Upper grid plate	14.2 ※1	43.2

※1 Assessment 60 years later after the commencement of operation
※2 Lower limit (BWR) for fracture toughness of radiation stainless steel from "Report on technological development for long-life plant" published by Japan Power Engineering and Inspection Corporation.

(2) Radiation-induced stress corrosion cracking of the upper grid plate

- **Regarding radiation induced stress corrosion cracking of upper grid plate,** estimated accumulated radiation amount of upper grid plate when main shock came exceeded to a small extent, but recorded value in PLM assessment (expansion coefficient for estimated missing stress of S2 seismic motion) is small enough that **even if considered seismic motion of main shock, it is extremely low possibility that it exceed fracture toughness**.

- **Also, just after the occurrence of main shock, reactor automatically shut down in the proper manner, and so even if the upper grid plate had been damaged, it was confirmed that there was no influence to insert control rod**

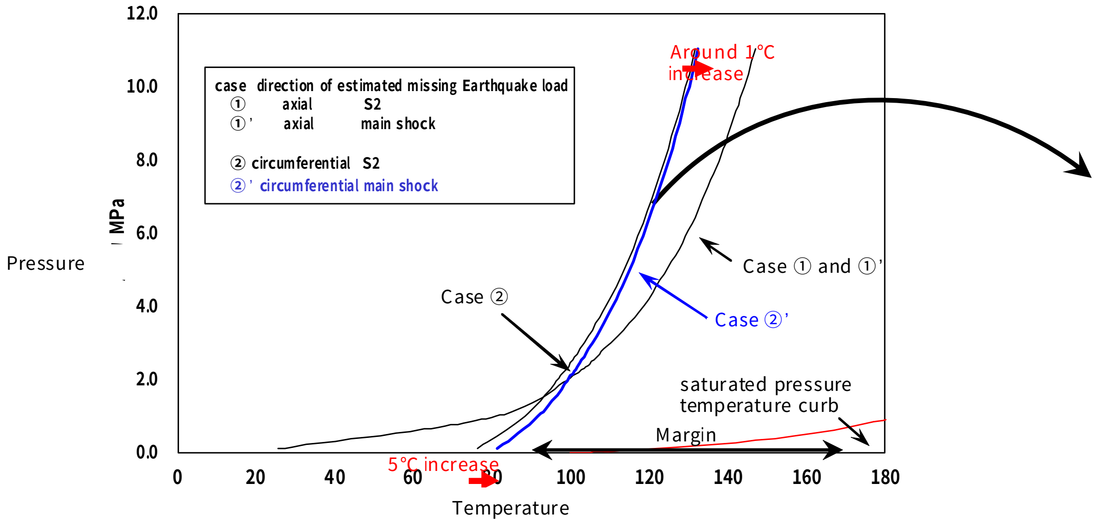
In respect of radiation induced stress cracking just after the occurrence of main shock, we evaluated that there was no influence to the soundness of facilities

（3） Neutron irradiation embrittlement of reactor pressure vessel

(Evaluation result of Reactor, Unit 1)

a. Regarding reactor pressure vessel（shell plate ）， estimate surface blemish on the conservative side, considering toughness lowering by neutron irradiation embrittlement, we calculated Fracture toughness （K_{IC}）
b. Considering axial direction of estimated missing, seismic movement does not work on expansion missing, we evaluated that occurrence of this earthquake does not influence.（case① and case①'matches（no difference from earth quake not consideration case））
c. Against calculation result of case ② of S2 earthquake load based on estimated missing of circumferential direction, we confirmed influence of main shock using case ②'that used main shock earthquake load
d. Pressure-temperature limited curb that considered influence of main shock（case ②'） does not cross-over saturated pressure temperature curb （operation curb of ＢＷＲ） indicates that we evaluated that when main shock was occurred, in respect of the neutron irradiation embrittlement, there was no influence to the soundness of facilities.

⇒Similar to Reactor Unit 1, at Reactors Units 2 & 3, there was sufficient room against the saturated pressure temperature curb. As such, we determined that there was no infulence to the soundness of facilities by the neutron irradiation embrittlement.



Unit	Temperature change at operating pressure
Unit 1	Approx 1℃
Unit 2	Approx 1℃
Unit 3	Approx 1℃

Pressure-Temperature limitation At Reactor Pressure Vessel(60 years)

（3） Neutron irradiation embrittlement of reactor pressure vessel

- **Sufficient margin has been confirmed for the seismic movement of the Earthquake, in addition to the existing evaluation result for the neutron irradiation embrittlement of reactor pressure vessel (at the time of 60 years of operation) and the virtual flaw in circumferential direction (depth “0.25t” and length “1.5t” of flaw for thickness “t” of the shell plate).**
- **There is no effect at the time of the Earthquake for the virtual flaw in axial direction, since the seismic movement does not act to extend the flaw.**

In terms of neutron embrittlement, it is concluded that there was no influence on the soundness of facility right after the occurrence of the Earthquake

Summary of evaluation result considering seismic movement of the Earthquake

Evaluation results are stated as follows, for the items for possible influence of age-related degradation right after the Earthquake, on the age-related degradation events for which the evaluation of deterioration development is required in addition to the daily maintenance activities.

（1） **Fatigue cracking of Unit 2 Reactor Pressure Vessel (feed water nozzle and Unit 3 Primary Containment Vessel (penetration bellows of feed water line)**

（2） **Radiation induced stress corrosion cracking of upper grid plate**

In terms of age-related degradation, it is concluded that there was no influence on the soundness of facility right after the occurrence of the Earthquake, since the influence of seismic movement was very small in the previous evaluation and it is quite unlikely to exceed the tolerance in the evaluation using the seismic movement of the Earthquake.

（3） **Neutron irradiation embrittlement of reactor pressure vessel**

In terms of age-related degradation, it is concluded that there was no influence on the soundness of facility, as the result of evaluation using the seismic movement of the Earthquake.

２．Influence of Age-related Degradation Immediately After the Tohoku-Chihou-Taiheiyou-Oki Earthquake

（１）verification of each event of Age-related Degradation

（２）PLM earthquake-proof safety evaluation based on result of earthquake response analysis of major equipments using observed record of the Tohoku-Chihou-Taiheiyou-Oki Earthquake

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Seismic movement for evaluation

<table>
<tr><th colspan="2" rowspan="2">Item</th><th colspan="2">Amplitude of maximum acceleration (free surface of the base stratum)</th><th rowspan="2">Remarks</th></tr>
<tr><th>Horizontal direction</th><th>Vertical direction</th></tr>
<tr><td colspan="2">Seismic movement of the Earthquake</td><td colspan="3">Evaluation using measured record
・For shear force and moment for the evaluation of core support structure, reactor pressure vessel, and primary containment vessel, they exceed the load by the design basis earthquake ground motion Ss.
・For floor response spectrum, although most of the results are below the those by the design basis earthquake ground motion Ss, there are some parts that the results of the Earthquake exceeding them, in the certain range of cycle (approx. from 0.2 to 0.3 seconds).</td></tr>
<tr><td rowspan="2">The previous seismic guideline</td><td>S_1</td><td>180 Gal</td><td>—</td><td>—</td></tr>
<tr><td>S_2</td><td>270 Gal
(Other than epicentral earthquake)
370 Gal
(Epicentral earthquake)</td><td>—</td><td>—</td></tr>
</table>

Major equipment and structure to be evaluated

Equipment・Structure	Evaluation part
Reactor Building	Seismic Wall
Reactor Pressure Vessel	Foundation Bolt
Primary Containment Vessel	Dry Well
Shut Down Cooling System Pump	Foundation Bolt
Core Support Structure (Structure inside of reactor)	Shroud Support
Main Steam Line	Main Line
Shut Down Cooling System Line	Main Line
Control Rod	Control Rod*

＊：Dynamic function evaluation (Insertability)

Evaluation result of major equipment (Unit 2)

Section	Evaluation equipment	Evaluation part	Stress classification	Calculated value (MPa)	Evaluation criteria*1 (MPa)	Evaluation method*2
Blocking	Core support structure	Shroud support	Axis compress stress	122	300	B
Cooling	Shutdown cooling system pump	Foundation bolt	Shear stress	45	185	B
	Shutdown cooling system line	Main line	Primary stress	87	315	B
Locking in	Reactor pressure vessel	Foundation bolt	Stretching stress	29	222	B
	Main steam line	Main line	Primary stress	208	360	B
	Primary containment vessel	Drywell	Envelope + bending stress	87	278*3	B

*1:Acceptable value for the condition D stated in "Standards for nuclear facility for power generation, Design and construction standard JSME S NC1-2005" (corresponding to acceptable stress condition $IV_A S$ stated in "Technical guideline for seismic design of nuclear power station JEAG4601 supplement-1984")

*2: A: Simplified evaluation, B: Detailed evaluation

*3: Evaluation criteria for the temperature during normal operation, since facility was under normal operation at the time of the earthquake

Section	Evaluation equipment	Unit	Calculated value	Evaluation criteria
Blocking	Control rod (insertability)	Relative displacement of fuel assembly (mm)	33.2	40.0

Evaluation result of major equipment (Unit 3)

Section	Evaluation equipment	Evaluation part	Stress classification	Calculated value (MPa)	Evaluation criteria*1 (MPa)	Evaluation method*2
Blocking	Core support structure	Shroud support	Axis compress stress	100	300	B
Cooling	Shutdown cooling system pump	Foundation bolt	Shear stress	42	185	B
	Shutdown cooling system line	Main line	Primary stress	269	363	B
Locking in	Reactor pressure vessel	Foundation bolt	Stretching stress	50	222	B
	Main steam line	Main line	Primary stress	151	378	B
	Primary containment vessel	Drywell	Envelope + bending stress	158	278*3	B

*1:Acceptable value for the condition D stated in "Standards for nuclear facility for power generation, Design and construction standard JSME S NC1-2005" (corresponding to acceptable stress condition $IV_A S$ stated in "Technical guideline for seismic design of nuclear power station JEAG4601 supplement-1984")

*2: A: Simplified evaluation, B: Detailed evaluation

*3: Evaluation criteria for the temperature during normal operation, since facility was under normal operation at the time of the earthquake

Section	Evaluation equipment	Unit	Calculated value	Evaluation criteria
Blocking	Control rod (insertability)	Relative displacement of fuel assembly (mm)	24.1	40.0

Major equipment and building to be evaluated and supposed events by age-related degradation

下表は２，３号炉で差異無し

Report on earthquake resistance reflecting seismic motion of main shock		Supposed events by age-related degradation in the technical evaluation of high aging management
Equipment, Building	Evaluated part	
Reactor Building	Seismic Wall	Strength degradation
		Shielding degradation
Reactor Pressure Vessel	Foundation Bolt	Uniform corrosion
Primary Containment Vessel	Dry Well	Uniform corrosion
Shut Down Cooling System Pump	Foundation Bolt*	**Uniform corrosion**
core support structure（structure in reactor）	shroud support	**fatigue crack**
		grain-boundary-type stress corrosion cracking
shut down cooling system pipe	Unit of pie	**fatigue crack**
main steam line pipe	Unit of pipe	**fatigue crack**
		flow accelerated corrosion，liquid droplet impact erosion
control rod	control rod	radiation inducement type stress corrosion cracking , grain-boundary-type stress corrosion cracking，toughness decrease

*The motor mounting bolts are with anti-corrosive coat. If the coat is sound, the possibility of corrosion is small. Also, as we did not observe significant corrosion in the past inspections, the foundation bolt that could have general corrosion is chosen as the subject.

The event which is affected by the principal earthquake significantly with the vibration response characteristics or structural & strength characteristics for the target facilities

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The reason for the conclusion that there was no influence of age-related degradation after the earthquake (1/3)

<table>
<tr><th colspan="2">Earthquake resistance report (seismic motion of main shock)</th><th rowspan="2">Supposed Age-related degradation</th><th rowspan="2">The reason for the conclusion</th></tr>
<tr><th>Equipment, Building</th><th>Evaluated part</th></tr>
<tr><td rowspan="2">Reactor Building</td><td rowspan="2">Seismic Wall</td><td>Strength degradation</td><td rowspan="2">・As for the evaluation on the earthquake resistance of concrete in Reactor Building, the robustness at the time of the earthquake was confirmed under the design conditions (age-related degradation was not considered)
・Meanwhile, as for the age-related degradation of concrete, it was confirmed by analyses and tests that there is no impact on the strength degradation of concrete.
・Therefore, regarding the influence of age-related degradation of concrete, it was concluded that the robustness at he time of the earthquake was confirmed following the evaluation result under the design conditions.</td></tr>
<tr><td>Shielding degradation</td></tr>
<tr><td>Reactor Pressure Vessel</td><td>Foundation Bolt</td><td>Uniform Corrosion</td><td>・As for the evaluation on the earthquake resistance of foundation bolts in Pressure Vessel, the robustness at the time of the earthquake was confirmed under the design conditions.
・Meanwhile, the exposed part of the foundation bolts is surrounded by nitrogen gas atmosphere in normal operation, so that it is less likely to be corroded. It was also confirmed by visual inspection that there is no significant corrosion so far.
・Therefore, regarding the influence of age-related degradation of foundation bolts, it was concluded that the robustness at he time of the earthquake was confirmed following the evaluation result under the design conditions.</td></tr>
</table>

The reason for the conclusion that there was no influence of age-related degradation after the earthquake (2/3)

Earthquake resistance report (seismic motion of main shock)		Supposed Age-related degradation	The reason for the conclusion
Equipment, Building	Evaluated part		
Primary Containment Vessel (PCV)	Dry Well	Uniform Corrosion	• We confirmed the soundness of the seismic assessment of the inner and outer surface of the drywell (upper mirror, cylindrical shell, spherical shell) at the time of the earthquake occurrence by the designed condition. • On the other hand, the assumed aged deterioration phenomenon at drywell are unlikely to be corrosion outbreak by painting the prevention of rust painting (synthetic resin system paint) and confirming that there was not deterioration meaningful until now in the result of the visual inspection. • Therefore, we judge that we can confirm the soundness of the drywell concerning the influence of the aged deterioration phenomenon at the time of the earthquake occurrence by the result of the evaluation of the designed condition.
core support structure (structure in reactor)	shroud support	grain-boundary-type stress corrosion cracking	• We confirmed the soundness of the seismic assessment of the shroud support at the time of the earthquake occurrence by the designed condition. • On the other hand, we assumed the occurrence and progress of the aged deterioration phenomenon at the shroud support by maintenance standard etc and inspected it systematically based on the evaluation of the seismic load. • Therefore, we judge that we can confirm the soundness of the shroud support concerning the influence of the aged deterioration phenomenon at the time of the earthquake occurrence by the result of the evaluation of the designed condition.

Reasons of judgment of no impact by the age-related degradation, at the time just after the earthquake occurred(3/3)

Seismic analysis report（earthquake motion by the earthquake）		Possible events by the age-related degradation	Reasons of the judgment
Equipment/Structures	Region of evaluation		
Main steam piping	Main body of the piping	Flow Accelerated Corrosion（FAC），Liquid Dropment Imprigement Erosion （LDI）	・Regarding seismic analysis on main steam piping, at design condition, seismic health was confirmed ・On the other hand, as for FAC, in the Rules on Pipe Wall Thinning Management by JSME (Japan Society of Mechanical Engineers), it is stated that the system is considered to have less incidence of FAC, considering environmental conditions of its inside fluid（dry steam, dissolved oxygen concentration, etc.）(FAC-1). In addition, as for LDI, in the Rules on Pipe Wall Thinning Management, it is stated that the system is considered to have small possibility of occurrence of LDI, considering environmental conditions of its inside fluid and elements of piping arrangement（valves, orifices, etc.）. ・Therefore, as for impact on the main stream piping by the age-related degradation, following the evaluation result at design condition, **we judged that we confirmed its health at the time when the earthquake occurred.**
Control rods	Control rods	Irradiation Assisted Stress Corrosion Cracking，Intergranular Stress Corrosion Cracking, Toughness Degradation	・Control rods had been replaced following an operation standard set by accumulated irradiance of thermal neutrons, and we had been confirmed that no problems on performance occurred through inspections. Therefore, **we judged that the possibility of the age-related degradation to cause significant impact to health of control rods at the time when the earthquake occurred is quite small.** ・In addition, we confirmed by the records of the Main Control Room, that all control rods were inserted at the time when the earthquake occurred.

（1）General corrosion of the foundation bolts of the pumps of Residual Heat Removal System (RHR)

Result of the analysis of the general corrosion of the foundation bolts of the pumps of Unit 2 RHR

Objects of the analysis	Seismic load	Shear Stress [MPa]		Allowable stress [MPa]
		Without corrosion	With corrosion	
Foundation bolts of the pumps of RHR	Earthquake motion of the earthquake	34	36	202

Result of the analysis of the general corrosion of the foundation bolts of the pumps of Unit 3 RHR

Objects of the analysis	Seismic load	Shear Stress [MPa]		Allowable stress [MPa]
		Without corrosion	With corrosion	
Foundation bolts of the pumps of RHR	Earthquake motion of the earthquake	23	24	202

（１）General corrosion of the foundation bolts of the pumps of Residual Heat Removal System (RHR)

- As the same as PLM report, **Analysis was conducted regarding the foundation bolts, assuming corrosion of them** by multiplying decrease ratio (3.9%) of sectional area, **with consideration of corrosion contents in 60 years (0.3mm)**
- Shear stress with assumption of corrosion content in 60 years **was confirmed to be lower enough than the allowable stress**

We judged that there was no impact by the assumed general corrosion on the foundation bolts of the pumps of the Residual Heat Removal System of Unit 2 and Unit3, at the time just after the earthquake occurred.

（2）Fatigue cracks of the shroud support structures inside the reactors

Result of the analysis of fatigue evaluation of the structures inside the reactors of Unit 2
（Numbers written in the PLM evaluation document）

Objects of the analysis	Cumulative usage factor based on number of operation times	Cumulative usage factor by motion of the earthquake (S_2 earthquake Motion)	Total （Allowable value is less than 1）
Shroud Supports	0. 323	0.000	0.323

Result of the analysis of fatigue evaluation of the structures inside the reactors of Unit 3
（Numbers written in the PLM evaluation document

Objects of the analysis	Cumulative usage factor based on number of operation times	Cumulative usage factor by motion of the earthquake (S_2 earthquake Motion)	Total （Allowable value is less than 1）
Shroud Supports	0.157	0.001	0.158

（2）Fatigue cracks of the shroud support structures inside the reactors

- If the ground motion changed, it only affects the calculation result of the cumulative usage factor by ground motion.
- The values (Cumulative usage factor by S2 ground motion) previously analyzed and written in the PLM evaluation document are small enough, therefore, it can be judged that the values does not exceed the allowable value even considering the ground motion.

It is concluded that there was no anticipated influence caused by fatigue crack on shroud support structures inside the reactors right after the occurrence of the earthquake.

（3） Fatigue cracks of pipes in Residual Heat Removal System (RHR)

For the evaluation result on the Primary Loop Recirculation System, which is usually in operation with higher pressure inside and higher temperature than RHR, the influence reflecting the change of motion ground instructed by new guideline of quake resistance was analyzed (the Primary Loop Recirculation System is connected with the Shut Down Cooing System. The evaluation condition is severer.).

Result of the analysis of fatigue evaluation of pipes in Primary Loop Recirculation System of Unit 2 （Numbers written in the PLM evaluation document）

Objects of the analysis	Cumulative usage factor based on number of operation times	Cumulative usage factor by motion of the earthquake (S_2 earthquake Motion)	Total （Allowable value is less than 1 ）
Primary Loop Recirculation System	0. 260	0.000	0.260

Result of the analysis of fatigue evaluation of pipes in Primary Loop Recirculation System of Unit 3 （Numbers written in the PLM evaluation document）

Objects of the analysis	Cumulative usage factor based on number of operation times	Cumulative usage factor by motion of the earthquake (S_2 earthquake Motion)	Total （Allowable value is less than 1 ）
Primary Loop Recirculation System	0.337	0.000	0.337

（3） Fatigue cracks of pipes in Residual Heat Removal System (RHR)

- If the ground motion changed, it only affects the calculation result of the cumulative usage factor by ground motion.
- The values (Cumulative usage factor by S2 ground motion) previously analyzed and written in the PLM evaluation document are small enough, therefore, it can be judged that the values does not exceed the allowable value even considering the ground motion.

It is concluded that there was no anticipated influence caused by fatigue crack on pipes in RHR of Unit 2 and Unit 3 right after the occurrence of the earthquake.

（4）Fatigue Crack of Main Steam Line Pipe

Result of the analysis of fatigue evaluation of Main Steam Line Pipe of Unit 2

Objects of the analysis	Cumulative usage factor based on number of operation times	Cumulative usage factor by ground motion (S2 Ground Motion)	Total (Allowable value is less than 1)
Main Steam Line	0. 366	0.001	0.367

Result of the analysis of fatigue evaluation of Main Steam Line Pipe of Unit 3

Objects of the analysis	Cumulative usage factor based on number of operation times	Cumulative usage factor by ground motion (S2 Ground Motion)	Total (Allowable value is less than 1)
Main Steam Line	0. 099	0.002	0.101

（4） Fatigue Crack of Main Steam Line Pipe

- If the ground motion changed, it only affects the calculation result of the cumulative usage factor by ground motion.
- The values (Cumulative usage factor by S2 ground motion) previously analyzed and written in the PLM evaluation document are small enough, therefore, it can be judged that the values does not exceed the allowable value even considering the ground motion.

It is concluded that there was no anticipated influence caused by fatigue crack on pipes in Main steam Line of Unit 2 and Unit 3 right after the occurrence of the earthquake.

Summary of evaluation result considering ground motion of the Earthquake

Based on the quake's evaluation result on the designing condition of major facilities and structures, the evaluation results for the items for possible influence of age-related degradation are stated as follows.

（1） General corrosion of the foundation bolts of the pumps of Residual Heat Removal System (RHR) **In terms of age-related degradation, it is concluded that there was no influence on the soundness of facility, as the result of evaluation using the ground motionof the Earthquake.**

（2） Fatigue cracks of the shroud support structures inside the reactors

（3） Fatigue cracks of pipes in Residual Heat Removal System (RHR)

（4） Fatigue Crack of Main Steam Line Pipe

Existing evaluation shows that the impact by motion ground was far small. In addition, because the evaluation using the quake's motion ground shows that it is unlikely that the value would exceed the allowable value, it is concluded that there was no influence on the soundness of facility.