Main Text

1. We dismiss all the claims of the Plaintiffs.
2. The Plaintiffs shall bear all of the court costs.

Facts

I. Petitions by the parties

1. Plaintiffs' petitions (Objects of Claims)

(1) The Defendant shall not operate the nuclear power plant which they built pertaining to the Prime Minister's license on December 10, 1970, in Tsukahama District and Fujimaruhama District, Onagawa Town, Oshika County Miyagi-prefecture.

(2) The Defendant shall not build the nuclear reactor pertaining to the Minster of International Trade and Industry's license on February 28, 1989, in Tsukahama District and Fujimaruhama District, Onagawa Town, Oshika County, Miyagi-prefecture.

(3) The Plaintiffs shall bear all of the court costs.

2. Defendant

(Answers before merits)

(1) We dismiss all the claims of the Plaintiffs.

(2) The Plaintiffs shall bear all of the court costs.

(Answers of merits)

Same as the Main Text

II. Parties' assertions

1. Plaintiffs' assertions

Plaintiffs' assertions, besides what as follows, are as written in the Exhibit of Plaintiff's final brief.

(1) Regarding the parties

a. Plaintiffs

The Plaintiffs are the residents living in Onagawa Town, Oshika County and Ishinomaki City, Miyagi-prefecture, and all of them are living in a 20 km radius from the nuclear power plant (hereinafter called the "Nuclear Power Plant") which the Defendant is operating and building in Tsukahama District, Onagawa Town, Oshika County and

Fujimaruhama District, Miyagi-prefecture, and may suffer the effect of radiation that the Nuclear Power Plant releases during usual operation and at an accident of the Nuclear Power Plant.

b. Defendant

The Defendant is a company that carries on general electricity business over seven prefectures in Tohoku area[1].

(2) Nuclear Power Plant

a. Reactor of Unit No. 1 of the Nuclear Power Plant

The Defendant applied to the Prime Minister on May 30, 1970, for an installment license for the Reactor of Unit No. 1 of the Nuclear Power Plant and obtained a license on December 10, 1970.

The overview of the licensed Reactor of Unit No. 1 of the Nuclear Power Plant is as follows:

(i) Type: Enriched uranium, Light water-moderated, Light water-cooled, Boiling water
(ii) Thermal Output: Approximately 1,590 megawatts
(iii) Electrical Output: Approximately 524,000 kilowatts

Thereafter, the Defendant applied the six alteration licenses for the Reactor in June 1974, October 1978, July 1980, April 1983, June 1986, and July 1991 and obtained all the respective licenses, and, although implemented partial design change, the fundamental structure did not change.

The Defendant completed the construction of the Reactor of Unit No. 1 of the Nuclear Power Plant in October 1983 and started its operation.

b. Reactor of Unit No. 2 of the Nuclear Power Plant

The Defendant applied to the Minster of International Trade and Industry on April 18, 1987, for a installment license for the Reactor of Unit No. 2 of the Nuclear Power Plant and obtained the license on February 28, 1989.

The overview of the licensed Reactor of Unit No. 2 of the Nuclear Power Plant is as follows

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[1] North area of Japan.

(i) Type: Enriched uranium, Light water-moderated, Light water-cooled, Boiling water
(ii) Thermal Output: Approximately 2,440 megawatts
(iii) Electrical Output: Approximately 825,000 kilowatts

Thereafter, the Defendant applied the alteration license and obtained a license in July 1991, and although implemented partial design change, the fundamental structure did not change.

The Defendant started the construction of the Reactor of Unit No. 2 of the Nuclear Power Plant on August 1989 and intends to start its operation in July 1995.

(3) The Plaintiffs, based on the personal rights or environmental rights, claim the judgment specified in the Column 1 and 2 of objects of claims against the Defendant.

2. Defendant's assertions

Defendant's assertions, besides what as follows, are as described in the first and the second volumes of Defendant's Exhibit final brief.

Among (1) and (2) of the Plaintiffs' assertions, the Defendant denies that the Plaintiffs suffer the effect of radiation that the Nuclear Power Plant releases during usual operation and accident, and accepts the other facts, and the assertions in Plaintiffs' Exhibit final brief are respectively contestable.

3. In "Reasons," as premises for making judgement, we will indicate the gist of the assertions including the assertions not described in the respective final briefs of the Plaintiffs and the Defendant as required.

III. Evidences (omitted)

REASONS

Chapter I Parties

There is no denial between the parties: the Plaintiffs are the residents living in Onagawa Town, Oshika County and Ishinomaki City, Miyagi-prefecture, Japan, and all of them are living in a 20 km radius from the Nuclear Power Plant which the Defendant is operating and building in Tsukahama district, Onagawa City, Oshika-country and Fujimaruhama District, Miyagi-prefecture, the Defendant is a company that carries on general electricity business over seven prefectures in Tohoku area (Aomori-prefecture, Iwateprefecture, Akita-prefecture, Miyagi-prefecture, Yamagata-prefecture, Fukushima-prefecture, and Nigata-prefecture); the Defendant applied to the Prime Minister on May 30, 1970, for the installment license for the Reactor of Unit No. 1 of the Nuclear Power Plant and obtained the license on December 10, 1970; the Defendant applied the six alteration licenses for the Reactor in June 1974, October 1978, July 1980, April 1983, June 1986, and July 1991 and obtained all the respective licenses, and although implemented partial design change, the fundamental structure did not change; the Defendant completed the construction of the Reactor of Unit No. 1 of the Nuclear Power Plant in October 1983 and started its operation; the Defendant applied to the Minster of International Trade and Industry on April 18, 1987, for an installment license for the Reactor of Unit No. 2 of the Nuclear Power Plant and obtained the license on February 28, 1989; the Defendant applied the alteration permit and obtained the permit in July 1991, and implemented partial design change, the fundamental structure did not change; the Defendant started the construction of the Reactor of Unit No. 2 of the Nuclear Power Plant in August 1989 and intends to start its operation in July 1995.

Chapter II Evidence of injunction claim

Since the Plaintiffs seek injunction of the operation of the Reactor of Unit No. 1 of the Nuclear Power Plant and injunction of the construction of the Reactor of Unit No. 2 of the Nuclear Power Plant based on the personal rights or environmental rights while the Defendant seeks dismissal of the claims by arguing that the personal rights or environmental rights are not based on instituted acts and that the request of the injunctions based on the personal rights or environmental rights is not entitled to be protected, the we will judge this respect.

Generally, it is needless to say that life and body of any individuals are crucially important protected interests, and it should be said that personal rights that refer to the safety of life and body of any individuals have exclusivity in the same manner as real right, and it is reasonable to construe that any individuals whose life and body are, or are likely to be, infringed can claim injunction of infringement act against offender based on personal rights in order to reject ongoing infringement act or prevent infringement that is to arise in the future (see the judgment of the Grand Bench dated on June 11, 1986 by the Supreme Court of Japan in 1981 (e) No. 609, Keishu Vol. 40, No. 4, p. 872).

Therefore, since it is obvious that the claims for injunction of the operation of the Reactor of Unit No. 1 of the Nuclear Power Plant and injunction of the construction of the Reactor of Unit No. 2 of the Nuclear Power Plant under the personal rights have a qualification as the right to claim under the Code of Civil Procedure, it should be said that the claims are legitimate.

Furthermore, although the environmental rights are not under an express provision under institute arts as the Defendant points out, it is not possible to jump to a conclusion, if taking into account each specific case, that the scope of right holders who are subject to the right, the range of environment that is subject to the right, and the contents of the right are unclear, and since it cannot be said that the claims based on the environmental rights do not have a qualification as a subject to examination in a civil trial, it should be said that the claims are legitimate.

However, although it should be said that further investigation is required regarding whether the right can be approved as the right to petition under substantive laws, since the Plaintiffs' claims for the injunction under the environmental rights are fundamentally the same as the claims under the personal rights with regard to whether it is possible to say if the Nuclear Power Plant poses enough risk against the Plaintiffs' environment to approve the injunctions of operation or construction, in the following, we will judge the presence or absence of risks of the Nuclear Power Plant.

(3) Judgment by the Court

According to the confirmed facts, it can be said that the Plaintiffs' assertions are based on reasonable materials such as research and study or the like, but, on the other hand, considering that these materials have been questioned and criticized and so on respectively by public organizations and experts, we must say that it is difficult to assert, based on these materials, the absence of the threshold between radiation exposure and occurrence of late onset disorders or the like in a natural scientific sense.

However, according to the testimony by (evidence number. omitted) and a witness, Sadao Ichikawa, regarding the presence or absence of the threshold between radiation exposure and disorders occurred in human body, it is general to agree with acute disorder among physical disabilities, but, regarding late onset disorder or the like, it is confirmed that: (i) there are few views which find the threshold; on the other hand, (ii) based on a theoretical basis that the severity of disorder does not change by the amount of exposed radiation but has the nature of changing its occurrence frequency, there are views which assert that the clear evidence is gained from experimental results of plants and animals, statistical results of humans and others and that although the occurrence frequency (probability) of late onset disorder or the like lowers as dose lowers, any low doses cannot make the probability zero, that is, there is no such threshold; however, (iii) from these results or the like, although it is not possible to assert the absence of the threshold, it can be accepted that the view that the assertion should be assumed or should be hypothesized in light of radiation protection are general.

Presumably, since there is the situation that the relationship between exposure dose and occurrence of late onset disorder or the like in low dose region has not been sufficiently analyzed yet, we cannot help saying that it is difficult to assert this as a matter of natural scientific demonstration, but, on the other hand, considering from an aspect or the like of radiation protection that it is general to take the view which assumes or hypothesizes that there is no threshold (as stated afterward, ICRP takes this view and hypothesize the linear relationship without the threshold), in usual discussions over fact finding in civil trial, in such case, although it is not impossible to consider that the relationship which is proved to such extent should be treated as absent, in statistical survey or the like, available subjects are usually limited by nature, and also experiments of damage on human life and body are prohibited and experiments of plants and animals are under constraints, and it is nothing but forcing impossible to demand demonstration of such relationship to the extent that is usually required in civil trial, and it seems that such demonstration will not be successful semi-permanently, and, meanwhile, since the interests that should be protected is human life and body which are crucially important, abatement and injunction of infringement act against these interests allow of no delay. Furthermore, the history tells that reports and views by public organizations and experts are not what will not definitely be revised and changed. There are too many things that science has not explained, and, regarding human physiology, pathology, heredity, and other mechanisms in particular, we must say that what we, human beings, have already revealed are considerably few.

Based on This, radiation is hazardous to human life and body, and when we evaluate the artificial release of this, as a legal judgment, if there is a proof to such extent, it is appropriate to see the absence of the threshold as for the relationship between exposure dose and late onset disorders or the like in low dose region as what can be approved.

4. Meaning of safety in nuclear reactor facilities

In the Nuclear Reactor Facilities, although various measures are taken to prevent the release of radioactive substances into the environment, it is difficult to avoid the release of the certain amount of radioactive substances into environment as previously confirmed in 1. The relationship between exposure dose and late onset disorders in low dose region is unsolved in many parts but can be accepted as described at the end of (3).

Also, as confirmed afterwards, the Nuclear Reactor facilities take various measures for safety assurance at the stages of basic design, construction and operation, but, on the other hand, as far as reactor are artificial facilities, it is a self-evident axiom that one cannot assert that an accident will never occur.

Therefore, if the safety required of reactor facilities means no possibility of occurrence of disorders by radiation caused by radioactive substances released from reactor facilities, and if it is construed that the Nuclear Reactor Facilities are no longer safe and the claims for injunction of construction or operation under the personal rights or the like should be accepted when there is possibility of occurrence of disorders by radiation, it must be said that the construction and operation of reactor facilities are almost impossible.

If it is assumed that all materials, devices, facilities and others or economic activities performed in human society should cause zero infringement or have zero risk of infringement against human life and body, besides nuclear power plant, the existence of most modern conveniences of contemporary society, such as radiography, television, and luminous watch which release radiation, or thermal power plant, hydroelectric power plant, automobile, and aircraft, even in consideration apart from a matter of radiation, cannot help being denied.

It is beyond question that such conclusion is against the social standards, and as described in Chapter IX, in consideration of the need for the Nuclear Power Plant from an aspect of power demand and supply, the safety required of reactor facilities, based on the assumption that reactor facilities inevitably release radioactive substances, should be construed to reduce the release of radioactive substances as low as possible for preventing the appearance of potential risks and to keep the risk of disaster occurrence low enough to be ignored under the social standards in any case.

Also, although it is needless to say that the safety of human life and body is the critical interest which requires full respect, if the possibility of disorder occurrence caused by the radiation from radioactive substances during the operation of reactor facilities is low enough to be ignored under the social standards, it should be construed that it is not possible to say there is infringement of human life and body by operating reactor facilities and that the claim for injunction based on unlawful infringement against the personal rights or the like is denied.

(4) Judgment by the Court

As we confirmed and judged above, although it is confirmed that there is the view or the like followed the Plaintiffs' assertions that the threshold limit value of ICRP is unjust, there are criticisms from experts against these views respectively, and according to the above-mentioned facts about the ICRP's organization, nature, activity or the like, the ICRP has been investigating various kinds of study results even today, and it can be confirmed that the ICRP has made recommendations in consideration of the views followed the Plaintiffs' assertions and criticisms against these. Therefore, the views relating to the Plaintiffs' assertions are not sufficient to overthrow the above-mentioned finding that the ICRP's threshold limit value is still the most dominant and proper allowed value (effective dose equivalent limit) in the world.

Furthermore, given the ICRP's views, we must say that it is reasonable to apply such threshold limit value as a reference for keeping the risks of radiation exposure low enough to be ignored under the social standards is reasonable, as far as the value is used with the so-called ALARA Principles which state that unnecessary exposure should be prevented and radiation exposure should be as low as achievable.

Since as stated above, it is reasonable to determine that the radiation dose, which can be said to make the possibility of occurrence of disorder low enough to be ignored under the social standards, should be the effective dose equivalent limit 0.1 rem per year which is based on the above-mentioned Article 2 of the public notice prescribing the dose equivalent limit or the like. However, since this value should be used with the principles which state that unnecessary exposure should be prevented and radiation exposure should be as low as achievable, in the following judgment, we additionally judge whether the measures for reducing the public exposure dose below such dose value as much as possible are taken.

II. Judgment by the Court

We judge that, in injunction lawsuits against construction and operation of nuclear power plants based on the personal right or the like, it is construed, in accordance with the principle of general injunction lawsuits, that it is the plaintiffs who bear the burden of proof for the risk that the Nuclear Power Plant may lack in safety and the Plaintiffs may suffer.

Therefore, considering in line with the present case, the Plaintiffs should bear the burden of proof concerning (i) the generation of radioactive substances by operating the nuclear power plant, (ii) the possibility of discharge of such radioactive substances to the outside during usual operation and accident, (iii) the possibility of spread of such radioactive substances, (iv) the possibility that such radioactive substances may reach Plaintiffs' bodies, and (v) the possibility of damage occurrence by radiation originated from such radioactive substances.

On the other hand, as we previously confirmed, the Nuclear Power Plant uses fuel uranium 235, and the operation generates a large amount of highly toxic radioactive substances harmful to human body such as plutonium 239, and the Plaintiffs are living within a radius of 20km from the Nuclear Power Plant; accordingly, it can be said that all Plaintiffs are living in the region where the residents are assumed to receive direct and serious harm on their life and body and other due to disasters caused by accident or the like in the Nuclear Power Plant, and, also, it is difficult for the Nuclear Power Plant to avoid releasing a certain amount of radioactive substances into the environment during usual operation as we previously confirmed.

As above, considering that the Plaintiffs have already proved the points from previously-mentioned from (i) to (v) as required and that the Defendant holds all sources about the safety of the Nuclear Power Plant, regarding the safety of the Nuclear Power Plant, the Defendant shall firstly provide the reasonable evidence to prove no lack in the safety and shall prove by submitting necessary sources including confidential data, and if the Defendant does not fulfill to prove, it should be said that the Nuclear Power Plant is virtually assumed (presumed) to lack in the safety. Also, if the Defendant fulfill to prove the safety of the Nuclear Power Plant as required, such virtual assumption regarding a lack in the safety is denied, and it should be construed that the Plaintiffs shall submit further proof regarding a lack in the safety.

II. Methods of Safety Review by Nuclear Safety Commission

1. NSC's basic stance and review for safety review

According to (evidence number omitted), it is confirmed that: the safety review by the NSC (Atomic Energy Commission, prior to amendment by the Act No.86 of 1978. Hereinafter, when referring to "NSC", it also includes the Atomic Energy Commission prior to amendment by the Act No.86 of 1978 if not specified) has the basic policy that general public and workers should not suffer radiation injury during usual operation, earthquakes, equipment failure and other emergency situations and that safety of general public should be secured in the assumption of an accident; particularly, five items are examined: (i) location condition, (ii) safety design of reactor facilities, (iii) exposure evaluation during usual operation, (iv) review of each accident (analysis of unusual transients during operation and accident analysis), and (v) location evaluation; from the countermeasure aspect, the examined items can be systematically divided into three, and basic stance and review for each are confirmed as follows.

(1) Countermeasures for accident prevention of nuclear facilities

The basic stance for accident prevention to ensure management of radioactive substances is taking appropriate countermeasures based on the concept of defense in depth. In accordance with this basic stance, the following points are examined when evaluating basic design at the stage of reactor installment license.

a. Prevention of unusual situation

(i) Can fission reaction of fuel be controlled stably?

(ii) Does fuel have sufficient safety allowance so that the soundness (in other words, confinement function for radioactive substances which is expected to ensure safety) is not impaired by thermal, mechanical, and chemical effects?

(iii) Does pressure boundary have sufficient safety allowance so that the soundness is not impaired by technical and chemical effects?

(iv) Does equipment other than such have performance, strength or the like with sufficient safety allowance to prevent the unusual situation which impairs the soundness of fuel and pressure boundary?

b. Expansion of unusual situation and prevention of escalation to accident

(i) Can fuel, pressure boundary, and the facilities for ensuring the soundness of these, if slight unusual situation occurs, detect the unusual situation early and definitely in order to take necessary countermeasures?

(ii) If the absence of quick countermeasures against the unusual situation results in a significant impact on the soundness of fuel and pressure boundary, such as the case in which the unusual situation occurred in the facilities for ensuring the soundness of fuel and pressure boundary is so serious, are safety protection facilities, such as reactor emergency shutdown system, to be installed in order to prevent fuel and pressure boundary from being impaired?

(iii) Can all safety protection facilities and others installed to prevent damages on fuel and pressure boundary exert their functions definitely?

c. Prevention of unusual release of radioactive substances

(i) Are engineering protection devices, such as emergency core cooling system, container or the like, to be installed in order to prevent the unusual release of radioactive substances even in an assumption that includes fracture of any pipes that constitute pressure boundary?

(ii) Can all engineering protection devices installed to prevent the unusual release of radioactive substances exert their functions definitely?

(2) Countermeasures for reduction of radiation exposure during usual operation of reactor

facilities

The basic stance for management of radioactive substances and preservation of surrounding environment during usual reactor operation is that the public exposure of the radiation from reactor facilities should be below the permissible exposure dose specified in laws and regulation, and, beyond that, should be well below this based on the ALARA's stance. In accordance with this basic stance, the following points are examined when evaluating basic design at the stage of reactor installment license.

(i) During usual reactor operation, are any appropriate countermeasures taken in order to prevent the occurrence of radioactive substances in first cooling water as much as possible?

(ii) During usual reactor operation, are any appropriate countermeasures taken in order to manage and control the radioactive substances occurred in first cooling water?

(iii) During usual reactor operation, is it assumed that the radioactive substances released into surrounding environment are monitored appropriately?

(iv) As a result of above countermeasures being taken, is the public exposure of radiation caused by the radioactive substances released into environment during usual reactor operation and by the radioactive substances inside reactor facilities kept below the permissible exposure dose specified in laws and regulation (as stated previously, 0.5 rem per year prior to March 31, 1989, and 0.1 rem per year since April 1, 1989), and well below this based on the ALARA's stance as a matter of course?

(v) Also, during usual reactor operation, are any countermeasures to be taken in order to sufficiently reduce the public exposure of radiation caused by radioactive substances inside reactor facilities?

(3) Countermeasures for location condition of reactor facility

For reactor installment, reactor facilities shall be installed with sufficient safety with respect to natural location conditions, and, to ensure safety of the public even in assumption of accident which is unlikely to occur in reality, reactors shall be located at a sufficient distance from the public in relation to their engineering protection devices. In accordance with this basic stance, the following points are examined when evaluating basic design at the stage of installment license.

a. Natural location conditions

The natural location conditions to be considered include ground, earthquake, weather, marine phenomenon, or the like; among these, regarding ground and earthquake which are put emphasis in safety review for reactor facilities, the following items are examined intensively.

(i) Is there a possibility that the ground of reactor sites generate large-scale landslides and debris flows which may cause damages on reactor facilities? Among the ground at reactor sites, does the bearing ground of nuclear facilities have the sufficient ground bearing capacity to support the facilities, and is there a possibility of generating ground breakage or the like by earthquakes and differential settlements due to loads?

(ii) Are earthquakes which should be considered to occur in the future around reactor sites selected appropriately from the history of past earthquakes or the like? Is the basis ground motion for design on reactor site base set with sufficient safety allowance in careful consideration of possible impacts of these earthquakes on reactor sites?

(iii) For this established basis ground motion for design, from engineering and technical perspectives, is it possible to develop seismic design with sufficient safety allowance for reviewed reactor facilities?

b. Separation between reactor facilities and the public

(i) Assuming the occurrence of severe accident (severe accident which may occur in the worst case from technical perspectives in consideration of events around sites, characteristics of reactors, engineering protection devices, and others), if the residents remain there, is the range of distance where the public may be exposed to radiation hazard determined as non-residential area?

(ii) Assuming the occurrence of hypothetical accident (the accident which exceeds severe accident and is unlikely to occur from technical perspectives), is the range of distance outside the non-residential area where the public may be exposed to significant radiation hazard unless certain countermeasures are provided determined as low population zone?

(iii) Assuming the occurrence of hypothetical accident, is the site separated by distance from the dense population zone so that the integrated value of the general exposure dose (the total dose of exposure dose to each member in a group), is small enough to be accepted from perspectives of national genetic dose?

2. Evaluation for methods of safety review by NSC

Based on the above confirmed facts, it is possible to find that the safety review for reactor installment license is conducted from the perspectives of whether the measures are provided for reactor facility locations, structures and devices in basic design to make the public exposure dose caused by the radioactive substances released into surrounding environment during usual reactor operation below the dose equivalent limit (0.1 rem per year) which is based on the above-mentioned Article 2 of the public notice prescribing dose equivalent limits, to reduce the public exposure dose below such dose equivalent limit as much as possible, and also to make the possibility of occurrence of accidents low enough to be ignored under the social standards, and it can be said that the methods of such safety review are appropriate to decide whether the public exposure caused by reactor installment can be restrained within the allowable unit.

Accordingly, when we evaluate the countermeasures in basic design of the Nuclear Reactor Facilities, we shall do this in accordance with the methods of the safety review, and we will herein present the assessment of the Plaintiffs' assertions in relevant parts.

III. Countermeasure of Reactor Facility for prevention of accident

1. Accident preventive measures

Among the countermeasures in basic design of the Reactor Facilities, as countermeasures relating to accident prevention for reactor facilities, the Defendant asserts preventive measures for occurrence of unusual events, preventive measures for expansion of unusual situation, and preventive measures for unusual release of radioactive substances. According to (evidence number omitted), it is confirmed that each countermeasure is evaluated in the Safety Review, and the basic design of the Nuclear Plant Facilities can ensure the safety relating to these measures. Accordingly, at first, we will review the specific contents of the review, and then present the assessment of the Plaintiffs' assertions.

According to (evidence number omitted) and all objects of the argument, specific contents of the review in the Safety Review are as follows.

(1) Perspectives of review for accident preventive measures

Regarding the radioactive substances generated in accompany with reactor operation, it is important for ensuring safety not only to manage them definitely during usual operation but to prevent unusual situation occurrence such as damages and others on fuel rods and pressure boundaries as a basis, to securely prevent escalation to the situation where radioactive substances may be unusually released into environment even if these unusualities occur, and furthermore to prevent the results of unusual release of radioactive substances into environment even in case of emergency by installing safety protection devices and others. Therefore, reactor facilities shall take various accident preventive measures based on the so-called “Defense in Depth,” and the Safety Review evaluated whether the accident preventive measures are taken from such aspects.

(2) Preventive measures for occurrence of unusual situation

The Safety Review, as follows, confirmed whether the measures to prevent unusuality occurrence in fuel cladding pipe, pressure boundary or the like which leads to the unusual release of radioactive substances into the environment are to be taken for the Reactor Facilities, and, as a result, it is confirmed that such measures are taken.

a. Secure and stable control of nuclear fission reaction of fuel

The mechanism of nuclear power generation is as described in 1 of Chapter III, and, to keep the soundness of fuel cladding pipe, at first, it is fundamentally required to enable secure and stable control of nuclear fission reaction of fuel.

The Safety Review found that fuel uranium 235 used in the Nuclear Reactor Facilities is of low concentration (the fuel assembly average fissile plutonium enrichment of high-burn-up 8x8 fuel, compared to new type 8x8 fuel, is changed from average about 2.3 wt.% to 2.5 wt.% in initial loading fuel, and from average about 3.0 wt. % to 3.5 wt. % in exchanged fuel); the Reactor is a light water nuclear power reactor with rapid and unique negative reactivity feedback features by void and Doppler effects and others in the whole operation range; that is, there is no possibility of uncontrollable nuclear fission reaction of fuel because of the presence of unique self-controllability (the feature that nuclear fission reaction is suppressed when the increased rate of nuclear fission reaction makes the temperatures of fuels and cooling water rise) against nuclear fission reaction; the Reactor Facilities are equipped with the control devises for stable control of nuclear fission reaction. As a result, the Nuclear Reactor Facilities are confirmed to control nuclear fission reaction securely and stably.

b. Soundness of fuel cladding pipe

The Safety Review confirmed that the fuel cladding pipe are designed with allowance in order to prevent damages on the pipe and its soundness as follows.

First, when a thermal output of fuel rod exceeds cooling capabilities of cooling water, heat removal is not provided sufficiently through fuel cladding pipe, and if boiling transition (the condition where the surface of fuel cladding pipe is covered by steam) occurs, fuel cladding pipe may be burned out. It is confirmed for the Reactor of Unit No. 1 that the value of minimal critical power ration (MCPR) is kept more than 1.23 and does not go below the permissible limit value, 1.06, which may cause burning of fuel cladding pipe during usual operation, and for the Reactor of Unit No. 2 that the permissible limit value does not go below 1.07.

Next, when a linear power density (output per unit length) of fuel rod increases, the gap between fuel pallet and fuel cladding pipe is lost due to the relative difference in thermal expansion between fuel pellet and fuel cladding pipe, and the fuel cladding pipe may be mechanically damaged. It is confirmed that a linear power density during usual operation of the Reactor Facilities is suppressed to 75 kilowatt per meter which may cause damages on the fuel cladding pipe by restricting to lower than 44 kilowatt per meter during usual operation.

Also, due to internal pressure from gaseous fission products and others leaching from fuel pellets and external pressure or the like from cooling water, fuel cladding pipe may be mechanically damaged. It is confirmed that the fuel cladding pipe used in the Reactor Facilities is designed with sufficient strength in consideration of stress cycle, fatigue limit, hydrogeneration of fuel cladding pipe, fretting corrosion, pellet-clad interaction or the like.

Furthermore, fuel cladding pipe may be damaged due to chemical corrosion caused by impurities in cooling water and others. It is confirmed that the fuel cladding pipe used in the Reactor Facilities has a special alloy (zircalloy II) with excellent corrosion resistance.

As a result of these findings, it is confirmed that the fuel cladding pipe used in the Reactor Facilities has allowance by which the soundness is not impaired by thermal, technical, and chemical impacts.

c. Soundness of pressure boundary

In the Safety Review, it is found that the pressure boundary is designed with allowance in order to prevent damages on the pressure boundary and keep its soundness as follows.

Firstly, when the pressure inside pressure vessel or the like becomes excessive, pressure boundary may be technically damaged. In the Reactor Facilities, it is confirmed that the pressure inside the pressure vessel can be almost fixed by the pressure control devise automatically and that the pressure boundary is designed with strength that has sufficient allowance against such pressure.

Secondary, if one uses a material with high brittle transition temperature, it may cause brittle fracture due to low temperature pressurization, and, particularly for pressure vessel, it may cause such brittle fracture in the state of high brittle transition temperature caused by constant exposure to neutron irradiation. In the Reactor Facilities, it is confirmed that the materials with high ductility in sufficient consideration to prevent brittle fracture are used, that, for the pressure vessels and others, they are kept at a temperature higher by 33 degrees Celsius or more than a brittle transition temperature of materials when pressurized, and that, particularly for the pressure vessel having problems of neutron irradiation, they are designed to install a monitor test piece on the inner wall to recognize the change of a brittle transition temperature.

Also, pressure boundary may be damaged due to chemical corrosion caused by impurities in cooling water and others. In the Reactor Facilities, it is confirmed that a stainless steel with excellent corrosion resistance is used as required and that the facilities are designed to perform an appropriate water quality management of cooling water such as monitoring the concentration of chlorine in cooling water, pH level, or the like which may cause corrosion.

Furthermore, it is confirmed that the equipment and pipes constituting the pressure boundary of the Reactor Facilities are designed to evaluate the soundness by inspection after the operation starts.

As a result of these findings, it is confirmed that the pressure boundary of the Reactor Facilities has allowance by which the soundness is not impaired by accident or the like.

d. Keeping reliability for equipment that may have an impact on soundness of fuel cladding pipe or the like

In the Safety Review, as the facilities that may have an impact on the soundness of the fuel cladding pipe, regarding the structure in reactor that comprises the core shroud or the like which supports and positions a fuel rod and provide the flow channel of cooling water, the reactor cooling system equipment to remove the heat generated from nuclear fission reaction of fuel, the reactor output control device or the like to control reactor output as follows, the reliability is confirmed as follows.

Firstly, it is confirmed that each facility used in the Reactor Facilities is designed to make its performance, strength, or the like have sufficient allowance so as not to have an impact on the soundness of the pressure boundary.

Secondly, in the Reactor Facilities, in order to prevent erroneous operations by operators, regarding the reactor cooling system equipment, the reactor output control device, or the like, it is confirmed that the measuring apparatus for pressure, temperature, flow or the like is installed to accurately grasp the state of such devices and that, regarding the reactor output control device, the interlock devise is installed to prevent extraction of the control rod even if the operator erroneously extracts the control rod in case that the number of neutrons in the reactor is larger than a determined value.

Also, in the Reactor Facilities, it is confirmed that, in order to continue the operation in case where the operation of the reactor deviates from a usual state, the automatic controller which automatically corrects such situation is installed, that, during usual operation for instance, the pressure controller which keeps the pressure in the pressure vessel constant by adjusting the stream control valve of the turbine inlet is installed, and that the water level controller which keeps a water level in the pressure vessel at the predetermined value by automatically controlling water supply flow is installed.

From the above results, it is confirmed that the facilities that may have an impact on the soundness of the fuel cladding pipe and pressure boundary in the Reactor Facilities assure the reliability which can prevent the occurrence of such unusual situations impairing the soundness.

(3) Preventive measures for expansion of unusual situation

In the Safety Review, as described above, although it is confirmed that the preventive measures for the occurrence of unusual situation are taken for the Reactor Facilities, in order to prepare for the case where unusual situation occurs nonetheless, as follows, it is evaluated whether the measures are to be taken to prevent the expansion of unusual situation and to prevent the escalation to the situation where radioactive substances may be unusually released into environment, and as a result, it is confirmed that such measures are taken.

a. Early and definite detection of unusual situation

If a minor unusual situation which may have an impact on fuel cladding pipes, pressure boundaries, and their soundness occurs, in order to take the requisite measures, it is necessary to detect the unusual situation early and definitely.

In the Safety Review, in the Reactor Facilities, it is confirmed that the measuring device which measures and monitors a radiation level in cooling water to detect damages on the fuel cladding pipe, the leak monitoring device which detects a leak of cooling water from the devices and others that constitute the pressure boundary, and the measuring device or the like which measures and monitors pressure, temperature, flow or the like of the reactor output and the reactor cooling system equipment are to be installed and that, in order to take necessary measures such as to stop the reactor in case unusual situation occurs is detected, the alarming device which immediately issues an alarm is to be installed. As a result, it is confirmed that the Reactor Facilities can detect the occurrence of such unusual situation early and definitely.

b. Installment of safety protection devices

If an unusual situation that occurs in the devices that may have an impact on the soundness of fuel cladding pipes and pressure boundaries is massive, a prompt measure shall be taken against the unusual situation.

In the Safety Review, it is confirmed that (i) a reactor emergency shutdown device, which prevents some kinds of unusual situations rise of thermal outputs and leads to a reactor scrum by inserting all control rods automatically and instantaneously if the occurrence of unusual situation in the reactor cooling system or the like causes a pressure in the pressure vessel to rise or a water level in the vessel to decrease unusually, is to be installed, that (ii) a reactor core isolation cooling system or the like, which keeps a water level in the pressure vessel by automatically supplying water to the vessel through the turbine drive pump and, in combination with the residual heat removal system, removes the decay heat or the like remaining in the core even after reactor shutdown and cools the fuel rod if water supply pumps and others stop after a scrum by a certain cause and a water level in the pressure vessel lowers due to the interruption of water supply into the pressure vessel, is to be installed, and that (iii) a release safety valve with main steam safety valve function, which prevents damages on the pressure boundary due to pressurizing by releasing the steam in the pressure boundary into the water of the pool in the suppression chamber and reducing the pressure if a pressure in the pressure vessel rises unusually, is to be installed. As a result, in the Reactor Facilities, it is confirmed that the devices (collectively the “Safety Protection Devices”) to take prompt and appropriate measures against the unusual events which may develop to have an impact on the soundness of fuel cladding pipes and pressure boundaries are to be installed.

c. Keeping reliability for safety protection devices

In the Safety Review, as follows, it is confirmed that all safety protection devices surely demonstrate the expected functions and the reliability is assured.

That is to say, it is confirmed that: (i) all safety protection devices installed in the Reactor Facilities are designed so as to obtain sufficient strength or the like; (ii) among safety protection devices, the reactor emergency shutdown device is designed to be capable of stopping the reactor by automatically inserting the control rod into the core in case that the power supply for such devices is lost by a certain cause, and the circuit for operating such device is designed to obtain redundancy and independence; also, even assuming that a control rod with maximum reactivity worth among all control rods is completely pulled out, it is designed to be capable of stopping the reactor by inserting the other control rods; furthermore, even assuming that the control rods cannot be inserted, a boric acid solution injection system capable of allowing the reactor to be in cold shutdown is installed; (iii) the reactor core isolation cooling system or the like is designed to be capable of removing the decay heat or the like and keeping a water level in the pressure vessel after a reactor shutdown by supplying cooling water through the operation of the turbine drive pump or the like by using a part of steams in the vessel which are generated from the decay heat of the core without using external power; (iv) as for the main steam system inside the container, the structure is simple, and the spring type which does not require any power and others is used for opening and closing operation; (v) in order to keep the reliability constantly, the safety protection devices are designed to be capable of implementing tests to check if the performance is continuingly secured after the operation starts.

d. Analysis evaluation for safety protection devices and others in unusual situation

As above, although it is confirmed that all safety protection devices of the Reactor Facilities keep the reliability, additionally in the Safety Review, as follows, the analysis evaluation assuming the occurrence of unusual transients during operation was conducted and the comprehensive validity of the designs of the safety protection devices and others were examined. That is, as such unusual transients, the analysis evaluation assumes the condition where the external disturbance exceeding usual reactor operation is imposed on the reactor by a single equipment failure, malfunction or single operational error that are expected in the lifetime of the nuclear installation and assumes the event which occurs with similar frequency of these and result in the condition where the reactor facility is not planned to operate; particularly, the safety review for the Reactor of Unit No. 1 concerning the report on November 16, 1970 respectively assumes (i) failure of recirculation pump, (ii) malfunction of recirculation flow control system, (iii) erroneous start of recirculation cooling water loop, (iv) failure of feedwater control system, (v) loss of supply water heating, (vi) loss of all feedwater flow, (vii) generator load break (quick close of turbine governing valve), (viii) turbine trip (close of main steam stop valve), (ix) close of main steam isolation valve, (x) failure of pressure controller, (xi) open of relief safety valve, (xii) control rod extraction at starting, (xiii) control rod extraction during output operation, and (xiv) loss of auxiliary power supply. The safety review for the Reactor of Unit No. 1 and No. 2 concerning the report on February 17, 1983 respectively assumes (i) control rod extraction at starting, (ii) control rod extraction during output operation, (iii) loss of external power source, (iv) loss of supply water heating, (v) erroneous start of recirculation stop loop, (vi) malfunction of recirculation flow control system, (vii) failure of recirculation pump, (viii) loss of load (generator load break and turbine trip), (ix) close of main steam isolation valve, (x) failure of feedwater control system, (xi) failure of pressure controller, and (xii) loss of all feedwater flow. The safety review for the Reactor of Unit No. 1 and No. 2 concerning the report on June 20, 1991 respectively assumes (i) unusual control rod extraction at staring, (ii) unusual control rod extraction during output operation, (iii) partial losses of reactor coolant flow, (iv) erroneous start of stop loop of reactor coolant system, (v) loss of external power source, (vi) loss of supply water heating, (vii) malfunction of reactor coolant flow control system, (viii) loss of load (generator load break and turbine trip), (ix) close of main steam isolation valve, (x) failure of feedwater control system, (xi) failure of reactor pressure control system, and (xii) all losses of feedwater flow, and the evaluation is conducted for these. When conducting the analysis evaluation for turbine trip (an event where a turbine is stopped by rapid shutdown of a steam stop valve at turbine inlet due to unusualities of turbine generator system and others, the pressure in a pressure vessel increases, and, as a result, the ratio of nuclear fission reaction of fuel rises and a fuel rod may be heated and damaged and also a pressure boundary may be damaged by the rise of pressure in the vessel), although the bypass valves installed in the bypass piping is expected to open automatically and to restrain the rise of the pressure in the vessel when the turbine trips, the severe conditions were set including the case that all of such bypass valves would not operate.

e. The Reactor of Unit No.1 currently uses the new type 8x8 fuel (this fact is confirmed by – evidence number omitted –). Among such analysis evaluation, the safety review concerning the report on February 17, 1983 which defines the use of the new type 8x8 fuel as a condition is more specifically as follows.

(i) As for MCPR, since it is assumed that the Reactor of Unit No. 1 operates while keeping it more than 1.23 during the period when the scrum curve for cycle early core is applied and more than 1.30 during the period when the scrum curve for cycle end core is applied respectively, it is confirmed that MCPR does not go below the permissible limit value, 1.06, even during “loss of supply water heating,” the severest transient at cycle early core and “loss of load,” the severest transient at cycle end core.

(ii) It is confirmed that, in the Reactor of Unit No. 1, the fuel retaining enthalpy is 31 calories per gram of uranium dioxide even in case of “control rod extraction at starting”, which is below the permissible limit value.

(iii) Regarding the soundness of the pressure boundary, it is confirmed that a reactor pressure is about 82.0 kg per square centimeter G at cycle early core and about 84.4 kg per square centimeter G at cycle end core, and thus below 1.1 times of the maximum use pressure.

f. The Reactor of Unit No. 1 uses high-burn-up 8x8 fuel as exchanged fuel, and the Reactor of Unit No. 2 will use high burn-up 8x8 fuel as initial loading and exchanged fuel (This fact is confirmed by – evidence number omitted –). The analysis evaluation in the safety review concerning the report on June 20, 1991 which required the use of high-burn-up 8x8 fuel is more specifically as follows.

(i) As for MCPR, since it is assumed that the Reactor of Unit No. 1 operates while keeping it more than 1.26 during the period when the scrum curve for cycle early core is applied and more than 1.36 during the period when the scrum curve for cycle end core is applied respectively, it is confirmed that MCPR does not go below the permissible limit value, 1.06, even during “loss of supply water heating,” the severest transient at cycle early core and “loss of load (turbine trip and inoperation of turbine bypass valve),” the severest transient at cycle end core; also, since it is assumed that the Reactor of Unit No.2 operates while keeping it more than 1.23 during both periods when the scrum curve for cycle early core and when the scrum curve for cycle end core is applied, it is confirmed that MCPR does not go below the permissible limit value, 1.07, even during “loss of supply water heating,” the severest transient at both periods .

(ii) It is confirmed that the severest linear power density of fuel in case of “unusual control rod extraction during output operation” in the Reactor of Unit No. 1 and No. 2 is about 53 kilowatt per meter (surface heat flux is equal to 121% of rated value) and that it is less than the linear power density equivalent to 1 % plastic strain of fuel cladding pipe (surface heat flux is equal to 170% of rated value).

(iii) It is confirmed that, in case of “control rod extraction at starting,” the highest enthalpy is 32 calories per gram of uranium dioxide in the Reactor of Unit No. 1 and 23 calories per gram of uranium dioxide in the Reactor of Unit No. 2, both of which are below the permissible limit value.

(iv) It is confirmed that a pressure on the pressure boundary becomes the severest in case of “loss of load (turbine trip and inoperation of turbine bypass valve)” in the Reactor of Unit No. 1 and reaches about 86.4 kg per square centimeter (cycle end core), which is below 1.1 times of the maximum use pressure (92.8 kg per square centimeter); in the Reactor of Unit

No. 2, the severest pressure in case of “loss of load (turbine trip and inoperation of turbine bypass valve)” reaches 92.8 kg per square centimeter, which is below 1.1 times of the maximum use pressure (96.7 kg per square centimeter).

As a result of above findings, it is confirmed that the Reactor Facilities can assure the soundness of fuel cladding pipe and pressure boundary in case that unusual transients occur, and it is confirmed that the designs of safety protection devices or the like of the Reactor Facilities is appropriate from the overall standpoint.

(4) Preventive measures for unusual release of radioactive substances

As described above, although it is confirmed in the Safety Review that the Nuclear Reactor Facilities take the preventive measures for the occurrence of unusual events and expansion of unusual events, to prepare for the event where radioactive substances are unusually released into the environment, as follows, it is evaluated whether the measures will be taken to prevent the unusual release of radioactive substances into the environment, and as a result, it is proved that such measures are taken.

a. Installment of engineering safety equipment

In the Safety Review, as follows, it is confirmed that the installed engineering safety equipment can prevent the unusual release of radioactive substances into the environment even assuming the unusual situation such as fracture or the like of any pipes constituting the pressure boundary.

That is, it is confirmed that, the Reactor Facilities install (i) an ECCS constituted by high-pressure water injection system, core spray system, low-pressure water injection system and automatic release valve system in the Reactor of Unit No. 1 and constituted by high-pressure core spray system, low-pressure core spray system, low-pressure water injection system and automatic depressurization system in the Reactor of Unit No. 2 in order to inject enough amount of cooling water to prevent a serious damage on the fuel cladding pipe in case of LOCA, (ii) a container vessel which secures high airtightness (the design leakage rate is less than 0.5% per day) to shut the radioactive substances released from the pressure boundary, (iii) a container vessel cooling system device which cools and decompresses the container vessel and additionally washes the radioactive substances floating in such steams to assure the soundness in case that high-temperature steams and others are released from the pressure boundary, (iv) a flammable gas concentration control system which suppresses a concentration of hydrogen or oxygen and prevents it from reaching the inflammable limit (design was changed by the permit on October 3, 1978 in the Reactor of Unit No. 1), and (v) an emergency gas treatment system or the like constituted by a filter for removing radioactive substances (the iodine elimination efficiency is 95% or more in design) or the like which captures the radioactive substances inside a reactor building leaked from a container vessel. As a result, it is confirmed that the Reactor Facilities install the necessary engineering safety equipment to prepare for the occurrence of events where radioactive substances are unusually released into the environment.

b. Keeping reliability for engineering safety equipments

In the Safety Review, as follows, it is confirmed that such engineering safety equipment can exhibit the expected function, and the reliability is assured.

That is, it is confirmed that (i) the engineering safety equipment installed in the Reactor Facilities has sufficient strength or the like and is designed to regularly conduct a test and an inspection, (ii) in order for the ECCS to surely exhibit its function, mutually independent two or more systems operate in case of any fractures of small caliber pipes which constitute the pressure boundary, and these systems are designed to provide emergency power supplies such as diesel generator in case of the loss of external power supply, (iii) the containment vessel is designed to keep a minimum working temperature 17℃ higher than a brittle transition temperature, and an isolation valve is installed at the penetration portion of containment vessel of the pipe which is required to close in case of LOCA or the like, and (iv) the container vessel cooling system facility and emergency gas treatment system install two mutually independent systems and provide emergency power supplies such as diesel generator or the like in case of the loss of external power supply.

c. Analysis evaluation for engineering safety equipments

As described above, although it is confirmed that all engineering safety equipment of the Reactor Facilities or the like assure the reliability, in the Safety Review, additionally, as follows, the analysis evaluation (accident analysis) which dares to assume the occurrence of accidents causing unusual release of radioactive substances into the environment was conducted, and the comprehensive validity of the designs of the engineering safety equipment was examined. That is, as such accidents, unusual events, which exceed above-mentioned unusual transient during operation, should be assumed in terms of safety evaluation for reactor facilities despite its low frequency since there is the possibility of radiation release from reactor facilities in case the accidents occur; particularly, the safety review for the Reactor of Unit No. 1 concerning the report on November 16, 1970, respectively assumes (i) control rod fall accident, (ii) control rod loss accident, (iii) fuel handling accident, (iv) LOCA, and (v) fracture accident of main steam pipe; the safety review for the Reactor of Unit No. 1 concerning the report on February 17, 1983, respectively assumes (i) coolant recirculation pump shaft fixation accident, (ii) LOCA, (iii) breakage accident of radioactive gaseous waste treatment facility, (iv) fracture accident of main steam pipe, (v) fuel handling accident, and (vi) control rod fall accident; and the safety review for the Reactor of Unit No. 1 and No. 2 concerning the report on June 20, 1991, respectively assumes (i) LOCA, (ii) loss of reactor coolant flow, (iii) coolant recirculation pump shaft fixation accident, (iv) control rod fall accident, (v) breakage accident of radioactive gaseous waste treatment facility, (vi) main steam pipe fracture, and (vii) fall of fuel assembly; regarding these events, assuming a single failure (one of the engineering safety equipment that has the most severe evaluation results fails due to a single event (including a multiple failure that inevitably occur due to a single event) and the safety function of the equipment is not demonstrated), the analysis was conducted by setting a precondition which results in severe evaluation results. As a result, it is confirmed that the Reactor Facilities can prevent the unusual release of radioactive substances into the environment in case of the occurrence of such accident, and it is confirmed that the design of the engineering safety equipment of the Reactor Facilities or the like is appropriate from overall standpoint.

Among such assumed events, regarding LOCA, a representative example of the case where radioactive substances are released into a containment vessel, fracture accident of main steam pipe, a representative example of the case where radioactive substances are released outside a containment vessel directly, and control rod fall accident, a representative example of other cases, the details of the accidents and the preconditions of the analysis are specifically as follows.

(a) LOCA

If a coolant in the core is lost due to the damages on a pipe that constitutes a pressure boundary, the fuel cladding pipe may be damaged by overheating and oxidation by water-zirconium reaction; also, the coolant outflow from the damaged parts of the pipe into the containment vessel and hydrogen gas or the like generated from such reaction may increase a pressure in the containment vessel, which may generate damages on the containment vessel.

In the analysis evaluation for such LOCA, the severe conditions are respectively assumed: as a major fracture accident, the coolant recirculation suction side pipe, which has the greatest amount of coolant loss and is the most severe for cooling the core, suffers complete fracture instantaneously; as a small and medium fracture accident, the pressure boundary pipe at the rupture area where the maximum temperature of the fuel cladding pipe reaches the highest suffer complete fracture instantaneously among the small and medium pipe; during usual operation, the reactor does not exceed the rated output, but it operates at

105% of the rated output; as the loss of external power supply occurring simultaneously with the accident and as a single failure, among the ECCS, an engineering safety equipment that operates during such accident to prevent unusual release of radioactive substances whose inoperation brings the most severe result is the failure of injection valve at a low-pressure water injection system in case of a major fracture accident and the failure of a high-pressure water injection system in case of a small and medium fracture accident.

(b) Fracture accident of main steam pipe

If the main steam pipe is fractured and the coolant outflow from the damaged parts occur, the fuel cladding pipe may be overheated and damaged.

In the analysis evaluation for such fracture accident of main steam pipe, the severe conditions are respectively assumed: one of four main steam pipes suffer complete fracture instantaneously outside the containment vessel; during usual operation, the reactor does not exceed the rated output, but it operates at 105% of the rated output; the external power supply is lost simultaneously with the occurrence of accidents, the coolant recirculation system pump shut instantaneously, and heat removal from the fuel cladding pipe decreases due to rapid decrease in core flow; as a single failure, one of signal circuits for closing main steam isolation valve due to the main steam pipe large flow does not function.

(c) Control rod fall accident

When a reactor goes into critical state or close to critical state, if a control rod falls for some reason, a reactivity is rapidly added, and the fuel cladding pipe may be overheated and seriously damaged.

In the analysis evaluation for such control rod fall accident, the severe conditions are respectively assumed: regarding the initial state of a reactor, the initial and end stages of cycle, the critical state at low temperature, and the critical state at high temperature during standby are respectively considered; one control rod with a value of 0.015ΔK, the design reference of rodworth minimizer, falls at the maximum value (0.95 meter per second) of the drop speed limited by the falling speed limiter; regarding reactor scram, one control rod with the maximum reactivity value is fixed at all draw-out positions and not inserted; the reactor emergency shutdown system shuts the reactor only by the high neutron bundle signal equivalent to 120% of the rated output, and its operation delay is 0.09 seconds; if the rate of nuclear fission reaction of nuclear fuel increased, nuclear fission reaction is oppositely suppressed by the Doppler effect, coolant temperature effect, and void effect, but among such suppressing effects, only the Doppler effect acts; a single failure occurs during neutron flux high scrum.

d. As previously found, the Reactor of Unit No. 1 currently uses the new type 8x8 fuel. Among such analysis evaluation, the safety review concerning the report on February 17, 1983 which requires the use of the new type 8x8 fuel as a condition is more specifically as follows.

(i) It is confirmed that the selection of models and parameters used in the analysis and the calculation code for assumption and analysis of a single failure are valid as both follow the "Regulatory Guide for Reviewing Safety Assessment of Light Water Nuclear Power Reactor Facilities" (Atomic Energy Commission decision on September 29, 1978) and the "Regulatory Guide for Evaluating Emergency Core Cooling System Performance of Light Water Nuclear Power Reactor" (Atomic Energy Commission decision on July 20, 1981).

(ii) In any postulated accidents, it is confirmed that the core is not damaged significantly, and that a sufficient cooling can be provided.

(iii) In any postulated accidents, it is confirmed that the pressure on the pressure boundary is below 1.2 times of the maximum use pressure.

(iv) In any postulated accidents, it is confirmed that the pressure on the containment vessel is below 10.9 times of the design pressure.

(v) It is confirmed that, regarding the maximum exposure dose outside the site boundary in case of accidents, the gamma ray whole-body exposure dose is approximately 0.011 rem in case of "breakage accident of radioactive gas waste treatment facilities" while the pediatric thyroid exposure dose is approximately 1.4 rem in case of "fracture of main steam pipe," both of which do not pose particular risks of radiation exposure to the public.

(vi) Regarding the performance evaluation for the ECCS, as a result of the analysis, it is confirmed that the maximum value of maximum temperature of the fuel cladding pipe is approximately 973℃ and that the maximum oxidation degree of the fuel cladding pipe is approximately 0.53%.

e. As previously found, the Reactor of Unit No. 1 will use high-burn-up 8x8 fuel as exchanged fuel while the Reactor of Unit No. 2 will use it as initial loading and exchanged fuel. The analysis evaluation in the safety review concerning the report on June 20, 1991 which requires the use of high-burn-up 8x8 fuel as a condition is more specifically as follows.

(i) It is confirmed that the selection of models and parameters used in the analysis and the calculation code for the assumption and analysis of a single failure are valid as both follow the “Regulator Guide for Reviewing Safety Assessment of Light Water Nuclear Power Reactor Facilities” (Atomic Energy Commission decision on August 30, 1990) and the “Regulatory Guide for Evaluating Emergency Core Cooling System Performance of Light Water Nuclear Power Reactor.”

(ii) In any postulated accidents, it is confirmed that the core is not damaged significantly, and a sufficient cooling can be provided.

(iii) It is confirmed that, in case of “control rod fall,” the highest value of fuel enthalpy is approximately 205 calories per gram of uranium dioxide in the Reactor of Unit No. 1 and approximately 208 calories per gram of uranium dioxide in the Reactor of Unit No. 2, both of which do not exceed the permissible limit value (230 calories per gram of uranium dioxide).

(iv) It is confirmed that the pressure on the pressure boundary is, in case of “control rod fall” when the pressure becomes the severest, approximately 84 kg per square centimeter in the Reactor of Unit No. 1 and approximately 88 kg per square centimeter in the Reactor of Unit No. 2, both of which are below 1.2 times of the maximum use pressure (101.3 kg per square centimeter in the Reactor of Unit No. 1 and 105.5 kg per square centimeter in the Reactor of Unit No. 2).

(v) It is confirmed that the pressure on the containment vessel is, in case of “LOCA,” approximately 2.9 kg per square centimeter in the Reactor of Unit No. 1 and approximately 3.3 kg per square centimeter in the Reactor of Unit No. 2, both of which are below the maximum use pressure (4.35 kg per square centimeter in both Reactors of Unit No. 1 and No.2).

(vi) It is confirmed that, the effective dose equivalent reaches the highest in case of the “fracture of main steam pipe” in the Reactor of Unit No.1 and the value is approximately 0.13 millisieverts while it reaches the highest in case of “fall of fuel assembly” in the Reactor of Unit No. 2 and the value is approximately 0.035 millisieverts, both of which do not pose particular risks of radiation exposure to the public.

(vii) Therefore, it is confirmed that the unusual release of radioactive substances into the environment can be prevented in any postulated accident.

(5) Conclusion

According to the specific review from (1) to (4) confirmed above, it can be confirmed that the safety judgement concerning the accident preventive measures for the Reactor Facilities is conducted based on the reasonable grounds, and in consideration of such facts and the above-confirmed organization and characteristics of the NSC, it can be assumed that the Reactor Facilities can assure the safety concerning the accident preventive measures.

IV. Countermeasures for reduction of radiation exposure during usual operation of the Reactor Facilities

1. Measures for reduction of radiation exposure during usual operation

Among the countermeasures in the basic design of the Reactor Facilities, as countermeasures for reduction of radiation exposure during usual operation of reactor facilities, regarding the amount of radioactive substances released into the environment during usual operation of reactor facilities, the Defendant, firstly, asserts that the public exposure dose does not exceed the exposure dose under the law, and, secondly, the measures are taken to reduce lower than such permissible exposure dose as practicable as possible. According to – evidence number omitted – the Safety Review evaluates such measures respectively, and it is confirmed that the basic design of the Reactor Facilities can assure the safety for these measures.

Thus, firstly, the specific review will be evaluated and then the claims by the Plaintiffs will be judged.

According to – evidence number omitted – and the entire object of the oral argument, the specific review of the Safety Review is confirmed to be as follows.

(1) Suppression of release of radioactive substances into environment

a. Suppression of occurrence of radioactive substances in cooling water

As previously stated, the main radioactive substances accumulated in reactor facilities during usual operation include (i) the nuclear fission products generated in fuel cladding pipe by fission reaction of fuel and (ii) the activation products generated by neutron activation of corrosion products and others such as iron rust or the like that are caused by the corrosion of equipment, the inside surface of pipe or the like contacting the cooling water; in order to reduce public exposure dose during usual operation of reactor facilities, firstly, it should be possible to suppress the occurrence of radioactive substances in cooling water; therefore, the occurrence of these should be prevented as much as possible by confining the products in fuel cladding pipe regarding the fission product (i) and by managing the quality of cooling water, suppressing the occurrence of the corrosion products in cooling water, and removing the generated products and others regarding the activation product (ii).

To this point, in the Safety Review, firstly, as stated in the countermeasures for nuclear fission products and others, it is confirmed that the soundness of the fuel cladding pipe used in the Reactor Facilities is maintained in design; also, for activation products, it is confirmed that the facilities for water quality management such as the reactor coolant purification system, condensate demineralization device, and others are used to maintain the quality of the cooling water in a clean state that prevents corrosion and that the stainless steel resistant to corrosion is used as a main material.

As a result, it is confirmed that the Reactor Facilities can prevent the occurrence of radioactive substances in the cooling water.

b. Management of radioactive substances outside reactor cooling system

As previously stated, in nuclear power plants, despite the measures to suppress the occurrence of radioactive substances in the cooling water, it is not available to completely eliminate the possibility of the formation of pinholes on a fuel cladding pipe, and fission products and others may leak into the cooling water from these pinholes or the like; also it is difficult to completely prevent the corrosion of equipment, the inside surface of pipe and others contacting the cooling water; accordingly, since the generation of the small amount of activation products is unavoidable, the occurrence of the small amount of radioactive substances in the cooling water is inevitable; therefore, when radioactive substances in the cooling water appear outside the reactor cooling system, it is necessary to keep the release of radioactive substances as low as possible by proper management.

To this point, in the Safety Review, as follows, it is confirmed that the Reactor Facilities are equipped with the radioactive waste disposal equipment that can properly process radioactive substances corresponding to each form of gas, liquid, and solid, and that the release of radioactive substances into the environment can be kept as low as possible.

(a) Gaseous radioactive substances

As previously stated, there are three types of the gaseous radioactive substances generated in the Reactor Facilities: (i) the radioactive substance contained in the condenser air ejector exhaust gas that is continuously extracted from the condenser by the condenser air ejector to keep vacuum in the condenser during usual operation, (ii) the radioactive substance contained in the ventilation system exhaust that is discharged to ventilate the air in the reactor buildings and others, and (iii) the radioactive substance contained in the steam condenser vacuum pump exhaust gas discharged intermittently from the condenser by operating the vacuum pump used for evacuation of the condenser when the turbine is restarted in a relatively short time after the stop; these gaseous radioactive substances include rare gas, particulate radioactive substances, and others.

In the Safety Review, it is confirmed that: regarding the radioactive substances related to the continuous release of (i), the Reactor of Unit No. 1 is equipped with the radioactivity attenuation tube that attenuates noble gas, the activated carbon type rare gas hold-up system with a retention time of more than 40 hours for krypton and more than 27 days for xenon, the exhaust gas particulate filter that captures particulate radioactive substances, and the exhaust pipe having a height of 123 meters or the like for diffusing and diluting rare gases and others while the Reactor of Unit No. 2 is equipped with the activated carbon type rare gas hold-up system with a retention time of 24 hours for krypton and more than 18 days for xenon, the exhaust gas particulate filter that captures particulate radioactive substances, and the exhaust pipe having a height of 160 meters or the like for diffusing and diluting rare gases and others; regarding the radioactive substances related to the ventilation system exhaust of (ii) and the intermittent release of (iii), the exhaust gas particulate filter and the exhaust pipe are provided.

As a result, it is confirmed that the Reactor Facilities are equipped with a disposal equipment that can properly process the gaseous radioactive substances.

(b) Liquid radioactive substances

As previously stated, there are four types of the liquid radioactive substances generated in the Reactor Facilities: (i) the water used to cool some auxiliary pumps or the like, which is the equipment drain having relatively a high radioactive substance concentration, (ii) the waste water used in the reactor buildings and others, which is the floor drain having relatively a low radioactive substance concentration, (iii) the waste liquid generated by regeneration of ion exchange resin (a process that washes out impurities contained in the resin by chemical treatment and returns the resin to its original state) used in the desalting device such as condensate demineralization device, which is regeneration waste liquid or the like having relatively a high radioactive substance concentration, and (iv) the waste liquid generated by washing clothes of plant workers, hand washing or the like, which is the laundry drain having extremely a low radioactive substance concentration.

In the Safety Review, it is confirmed that: regarding the equipment drain of (i), the filtration apparatus for removing solid contents, the desalting device for removing ionic substances or the like are provided (The treated water is collected in the water liquid sample tank and reused as reactor cooling water or the like.); regarding the floor drain of (ii), the evaporation concentration equipment for distilling, the desalting device, and others are provided (In principle, the distilled water is collected in the floor drain sample tank after being desalted and then reused as reactor cooling water. The concentrated waste liquid remaining after distillation is solidified and processed as solid radioactive substances.); regarding the regeneration waste liquid of (iii), the neutralization tank, the evaporation concentration equipment and others are provided (The distilled water is processed by the equipment drain system of (i). The concentrated waste liquid is processed in the same way as the floor drain of (ii)); and regarding the laundry drain of (iv), the pretreatment apparatus for removing solid contents and the evaporation concentration equipment are provided (The distilled water is collected in the laundry drain sample tank and reused as washing water whenever possible. A portion of the distilled water is mixed and diluted with seawater for cooling the condenser and released into the environment. The concentrated waste liquid is processed in the same way as the floor drain of (ii)).

As a result, it is confirmed that the Reactor Facilities are equipped with the disposal equipment that can properly process the liquid radioactive substances.

(c) Solid radioactive substances

As previously stated, there are four types of the solid radioactive substances generated in the Reactor Facilities: (i) the used ion exchange resin such as desalting devices used in the process of cooling water purification and liquid waste treatment, (ii) the concentrated waste liquid generated by the evaporating and thickening treatment of floor drain, regeneration waste, and laundry drain, (iii) the laundry waste sludge generated by the pretreatment of laundry drain, and (iv) the solid waste such as a piece of cloth and wastepaper with radioactive substances attached by contacting cooling water during equipment inspections and repairs and used filters depleted for processing gaseous wastes.

In the Safety Review, it is confirmed that: regarding the used resins of (i) or the like, the used resin storage tank and the sedimentation layer to temporarily store and attenuate radioactivity, the incinerator, and the device that mixes with a solidifying agent and cans the drum are provided; regarding the concentrated waste liquid of (ii), the concentrated waste liquid storage tank to temporarily store and attenuate radioactivity, and the device that mixes with a solidifying agent and cans the drum are provided; regarding the laundry waste sludge of (iii), the sedimentation to temporarily store and attenuate radioactivity, the incinerator, and the device that mixes with a solidifying agent and cans the drum are provided; regarding the solid waste of (iv), the compression volume reduction apparatus, the incinerator, the device that mixes with a solidifying agent and cans the drum are provided; furthermore, such drums are preserved and stored in the solid waste storage facility; the waste gas generated from incineration is released from the exhaust port of the waste incineration facility after being processed by the ceramic filter and the high-performance particulate filter.

As a result, it is confirmed that the Reactor Facilities are equipped with a disposal equipment that can properly process the solid radioactive substances.

(2) Evaluation for public exposure dose

In the Safety Review, in the Reactor Facilities, as described above, it is confirmed that, since the measures to prevent the release of radioactive substances into the environment are taken, the radioactive substances released into the environment are limited to only a portion of the gaseous and liquid substances generated in the Reactor Facilities.

In addition, the public exposure dose caused by the radioactive substances released into these environment is evaluated as follows, and it is confirmed that the evaluation value does not exceed the permissible exposure dose for the public (0.5 rem per year until March 31, 1989, and 0.1 rem per year since April 1, 1989) and can be kept still lower as practical as possible.

a. For the following reasons, it is confirmed that the exposure dose assessment in the Safety Review can provide reasonable conclusions without conducting other forms of exposure assessment if the evaluation takes (i) external whole-body exposure to gamma ray of rare gas, (ii) internal thyroid exposure due to iodine intake, and (iii) internal whole-body exposure by intake of marine organisms.

(a) Among the radioactive substances released into the environment during usual operation, the one with the largest amount is a rare gas contained in gaseous waste. It is considered that gamma rays released from the rare gas diffused into the atmosphere may cause external whole-body exposure.

(b) Among the radioactive substances released into the environment during usual operation, although iodine discharge is smaller than rare gas, iodine is contained in gaseous waste and liquid waste, and after being released, it is concentrated by marine organisms and attached to leaf vegetables. Therefore, it is considered that, if iodine is taken into human body by the intake of marine organisms and leaf vegetables (including milk from cows that eat leaf vegetables), it is selectively concentrated into thyroid and may cause internal exposure to thyroid.

(c) Among the radioactive substances released into the environment during usual operation, iron, manganese, cobalt, and others are scarcely contained in gaseous waste but account for a large proportion in liquid waste. These elements are concentrated into marine organisms, and it is considered that, if these radioactive substances are taken into human body by the intake of marine organisms, they may cause internal whole-body exposure dose.

(d) The exposure dose from all other forms of exposure to radioactive substances released into the environment during usual operation of reactors is negligibly small compared to the dose from (a) to (c).

b. The radioactive substances to be evaluated for the public exposure dose during the usual operation of the Reactor Facilities are as follows, and it is confirmed that all of them are based on the results of the preceding nuclear reactors and others and thus confirmed reasonable.

(a) In the Safety Review, regarding the radioactive substances released during the usual operation of the Reactor of Unit No. 1, it is assumed that, in gaseous waste, radioactive rare gases are 38,000 Curies per year, iodine 131 is 2.3 Curies per year, and iodine 133 is 4.5 Curies per year while, in liquid waste, tritium is 100 Curies per year, and the radioactive liquid waste excluding tritium is 0.1 Curies per year. Also, regarding the radioactive substances released during usual operation of the Reactor of Unit No. 2, it is assumed that, in gaseous waste, radioactive rare gases are 32,000 Curies per year, iodine 131 is 0.55 Curies per year, and iodine 133 is 0.91 Curies per year while, in liquid waste, tritium is 100 Curies per year, and the radioactive liquid waste excluding tritium is 0.1 Curies per year, and by combining with the Reactor of Unit No. 1, it is assumed that, in gaseous waste, radioactive rare gases are 70,000 Curies per year, iodine 131 is 2.9 Curies per year, and iodine 133 is 5.4 Curies per year while, in liquid waste, tritium is 200 Curies per year and the radioactive liquid waste excluding tritium is 0.2 Curies per year.

(b) Among mentioned above, the grounds for the assumption of the rare gas in gaseous waste released during usual operation of the Reactor of Unit No. 1 are as follows.

Firstly, it is respectively assumed that (i) the amount of the rare gas contained in the condenser air ejector exhaust gas that is continuously extracted from the condenser by the condenser air ejector is 22,000 Curies per year (0.86 millicuries per second, annual operation rate of 80 %), (ii) the amount of the rare gas contained in the ventilation system exhaust is 7,600 Curies per year for the turbine buildings and 5,300 Curies per year for the radioactive waste disposal buildings, and (iii) the amount of the rare gas contained in the steam condenser vacuum pump exhaust gas discharged intermittently from the condenser by operating the vacuum pump is 3,800 Curies per year (750 Curies per discharge, 5 times per year).

Among these, the value of (i), 0.86 millicuries per year, is based on the assumption that the leak rate of all rare gas from the core fuel to the coolant is 0.3 Curies per second (attenuation conversion value of annual average 30 minutes), and this is assumed as the most severe condition in consideration to the operation results of the preceding nuclear power plants (The annual average value of the total leak rate of all rare gas is from 0.01 to 0.3 Curies per second (attenuation conversion value of annual average 30 minutes)). Also, the value of (ii) is calculated by using the average value of the ratio between the measured value of the ventilation system rare gas release rate during operation and the leak rate of all rare gas from the core fuel based on the operation results of the preceding reactors. The estimated value of the ventilation system of the reactor buildings and the waste treatment buildings is to be neglected in the calculation because, among the rare gas, the abundance ratio of nuclides except xenon 133 or the like is small. The value of (iii) is assumed that the discharge amount per vacuum pomp is 2,500 Curies and that the number of the operations is 5 times when the leak rate of all rare gas is 1 Curie per second based on the operation results of the preceding nuclear power plants.

In addition, regarding the grounds of the assumption for the rare gas in gaseous waste released during usual operation of the Reactor of Unit No. 2, they are similar except being assumed that the leak rate of all rare gas from the core fuel to the coolant is 0.3 Curies per second (attenuation conversion value of annual average 30 minutes), and the amount of the rare gas contained in the condenser air ejector exhaust gas that is continuously extracted from the condenser by the condenser air ejector is 1.0 millicurie per second.

(c) Among mentioned above, the grounds for the assumption of the iodine in gaseous waste released during usual operation of the Reactor of Unit No. 1 are as follows.

Firstly, it is respectively assumed that (i) the amount of the iodine contained in the condenser air ejector exhaust gas that is continuously extracted from the condenser by the condenser air ejector is 0 Curie per year, (ii) the amount of the iodine contained in the ventilation system exhaust is 1.6 Curies for iodine 131 and 4.3 Curies for iodine 132 during usual operation and 0.6 Curies for iodine 131 during periodical inspection, and (iii) the amount of the iodine contained in the steam condenser vacuum pump exhaust gas discharged intermittently from the condenser by operating the vacuum pump is 0.12 Curies for iodine 131 and 132 (0.024 Curies per discharge, 5 times per year).

Among these, for iodine 131 (half-life of 8.06 days) and iodine 133 (half-life of 2.08 hours), since the retention period by the activated carbon type rare gas hold-up system is longer than such half-life, it is confirmed that the release of (i) can be ignored. Additionally, based on the operation results of the preceding reactors, the value of (ii) during usual operation is calculated by using the leakage coefficient defined by the average value of the ratio between the measured value of the rate of the iodine release from the operating ventilation system and the iodine concentration in the coolant. Also, based on the operation results of the preceding reactors, the value of (ii) during periodical inspection is calculated by using the average value of the ratio between the measured value of iodine 131 and the leak rate of all rare gas from the core fuel before stop. Regarding iodine 133, it is neglected in the calculation because the amount of its release is considered to be smaller than that of iodine 131 due to the short half-life during periodical inspection. Furthermore, based on the operation results of the preceding nuclear power plants, the value of (iii) is assumed that the discharge amount of iodine 131 and iodine 133 per vacuum pomp, when the leak rate of all rare gas is 1 Curie per second, is 0.08 Curies and that the number of operations is 5 times.

In addition, regarding the grounds of the assumption for the iodine in gaseous waste released during usual operation of the Reactor of Unit No. 2, they are similar except assuming that the leak rate of all rare gas from the core fuel to the coolant is 0.3 Curies per second (attenuation conversion value of annual average 30 minutes).

(d) Among mentioned above, the ground for the assumption of the radioactive liquid waste in liquid waste released during usual operation of the Reactor of Unit No. 1 is as follows.

The discharge amount of tritium in liquid waste is assumed to be 100 Curies per year based on the operation results of the preceding nuclear power plants. Also, regarding the radioactive liquid substances except tritium, since floor drain is assumed to be 1,000 cubic meters per year (0.01 Curies per year) and laundry drain is 4,000 cubic meters per year (0.004 Curies per year), 0.1 Curies is assumed as a discharge control target.

Regarding the grounds of the assumption for the radioactive liquid waste in liquid waste released during usual operation of the Reactor of Unit No. 2, they are similar except of assuming that, as for the radioactive liquid substances other than tritium, floor drain and chemical waste liquid are 3,000 cubic meters per year (0.03 Curies per year) and that laundry drain is 2,600 cubic meters per year (0.0026 Curies per year).

c. The following conditions are assumed for the evaluation of public exposure dose during usual operation of the Reactor Facilities.

(a) The gaseous waste released into the environment from the Reactor Facilities is diffused and diluted in the atmosphere and reaches outside the Environmental Monitoring Area. In evaluating the external whole-body exposure dose by gamma rays released from the rare gas in gaseous waste and the thyroid exposure dose due to the iodine in gaseous waste, the actual measurement values from the meteorological observation on the Nuclear Power Plant site from November 1980 to October 1981 are used regarding the iodine released from the exhaust pipe into the environment and the diffusion and dilution of rare gas.

(b) In evaluating the external whole-body exposure dose by gamma rays released from the rare gas in gaseous waste, it is performed for the outside of the Environmental Monitoring Area in consideration of residential area or the like, and it is assumed that the dose at the point where such external whole-body exposure dose becomes the maximum in a year is defined as the evaluation value. In other words, such evaluation assumes a severe condition, in which a person remains on the boundary of a power plant site for a year, that never actually occurs.

(c) In evaluating the thyroid exposure dose due to the iodine in gaseous waste, respiration and intakes of leaf vegetables and milk are considered as the exposure routes, and the exposure dose is evaluated by assuming that: as for leaf vegetables, adults eat 100 grams a day, infants take 50 grams a day, and babies take 20 grams a day continuously for a year; as for milk, adults drink 200 grams a day, infants drink 500 grams a day, and babies drink 600 grams a day continuously for a year. Also, the evaluation of the exposure dose due to respiration is conducted at the point where the annual average concentration of iodine becomes the maximum outside the boundary of the site; additionally, the evaluation of the exposure dose due to the intakes of leaf vegetables and milk, among the places where leaf vegetables and farms used as fodders of cows actually exist, assumes that each person ingests the leaf vegetables and milk from the cows that eat the grass at the point where the annual average concentration of iodine becomes the maximum.

In other words, such evaluation assumes a severe condition, in which a person remains in an area with the maximum concentration for one year and the same person continues to ingest only leaf vegetables and milk at the point where the concentration becomes the highest among the points where leaf vegetables and the grass actually exist near the nuclear power plant, that never actually occurs.

(d) The liquid waste discharged from the Reactor Facilities into the seawater is diffused and diluted in the seawater and taken in by marine organisms and others, and the public is exposed by intake of it or the like. In evaluating internal whole-body exposure due to radioactive substances in liquid waste and thyroid exposure dose due to iodine in liquid waste, diffusion and dilution by seawater are taken into no consideration, and a concentration at the condenser cooling water drain port is determined as the concentration in the front sea area, and as for marine organisms, the evaluation of exposure dose is performed by respectively assuming that adults eat 200 grams of fish and 20 grams of invertebrate a day, and infants and babies eat one-half and one-fifth of the adults' values. In other words, such evaluation assumes a severe condition, in which marine organisms continue to live at the outlet of liquid waste into the seawater throughout a year and are collected and served for public consumption and the same person continues to ingest only it every day for a year, that never actually occurs.

(e) In evaluating internal whole-body exposure due to radioactive substances in liquid waste and thyroid exposure dose due to iodine in liquid waste, the concentration of radioactive substances in marine organisms is considered, and as for the degree of the concentration (concentration coefficient), the stable element concentration measurement value for the edible parts of marine organisms is widely quoted from the literature, and the degree shall be based on what is obtained by collectively calculating the representative values.

d. Based on the evaluation methods above, the maximum value of public exposure dose during usual operation of the Reactor Facilities is estimated, if only the Reactor of Unit No. 1 is operated, that the external whole-body exposure dose due to gamma ray of rare gas is approximately 0.63 millirem per year, that the internal exposure dose due to beta and gamma rays released from radioactive substances in liquid waste is approximately 0.04 millirem per year, and that the thyroid exposure dose due to beta and gamma rays released from iodine in gaseous and liquid waste that is taken in human body is approximately 1. 3 millirem per year; if the Reactors of Unit No. 1 and No.2 are operated, the external whole-body exposure dose due to gamma ray of rare gas is approximately 0.9 millirem per year, that the internal exposure dose due to beta and gamma rays released from radioactive substances in liquid waste is approximately 0.04 millirem per year, and that the thyroid exposure dose due to beta and gamma rays released from iodine in gaseous and liquid waste that is taken in human body is approximately 1. 5 millirem per year.

As a result, it is confirmed that the Reactor Facilities can keep the evaluation value of public exposure dose caused by radioactive substances released into the environment during usual operation much below the permissible exposure dose and maintain it to be still lower as practical as possible.

(3) Monitoring release amount of radioactive substances or the like

As previously stated, when releasing radioactive substances into the environment during usual operation of reactor facility, in order to confirm that the radioactive waste disposal equipment usually functions, it is necessary to install the equipment that can properly monitor the release amount, the exposure dose, and others after the discharge, and, in the Safety Review, it is confirmed as follows.

Firstly, it is respectively confirmed that: as for gaseous waste, a radiation monitor is installed in the exhaust pipe in order to continuously monitor the release amount of radioactive substances from the exhaust pipe into the environment; as for liquid waste, in order to check that the concentration of radioactive substances is sufficiently low before being released into the environment, the facilities that store them temporarily in the sample tank and then sample and measure the concentration of radioactive substances are installed, and a radiation monitor that can continuously monitor the release amount of the radioactive substances is installed at the drainage pipe connected to the cooling water drainage path of the condenser. Also, regarding the monitor of the radiation dose rate in the environment or the like, it is confirmed that the facilities that measure the radiation dose rate or the like, such as monitoring posts, are installed around the Reactor Facilities. As a result, in the Reactor Facilities, it is confirmed that the radiation control facilities that can properly monitor the release amount of radioactive substances released into the environment during usual operation, radiation dose and radiation concentration in the environment, and others are installed.

(4) Conclusion

According to the above-confirmed specific review from (1) to (3), it can be confirmed that the safety judgement relating to the exposure reducing countermeasures during usual operation of the Reactor Facilities in the Safety Review is based on the reasonable grounds, and in consideration to such fact and the previously-confirmed organization and characteristics of the Nuclear Safety Commission, if there is no additional claim and evidence regarding this point from the Plaintiffs, it can be said that the safety relating to the exposure reducing countermeasures during usual operation is presumed to be assured.

V. Countermeasures for location conditions of Reactor Facilities

1. Natural Location Conditions

The safety judgement related to the natural location conditions of reactor facilities shall be based on the comprehensive review which concerns whether the Reactor Facilities are designed and built to be safe from the engineering and technical perspectives in its basic design corresponding to the natural location conditions. In such review, the natural location conditions to be considered include ground, earthquake, weather, marine phenomenon or the like, and among these, ground and earthquake are particularly emphasized in the safety review of reactor facilities. In the following, since the Plaintiffs assert the dangers related to the natural location conditions of the Reactor Facilities with a focus on such points, we will mainly assess the issues of the ground and earthquakes.

As the countermeasures for the natural location conditions of the Reactor Facilities, the Plaintiffs assert that: firstly, as the site of the Nuclear Power Plant, the ground that has the necessary bearing capacity and no risk of causing ground breakage by earthquakes is selected; secondly, regarding the geological structure around the site of the Nuclear Power Plant, the sufficient investigations and examinations are conducted; thirdly, based on the investigations into such geological structure, the aseismic design is sufficiently performed on the Nuclear Power Plant. According to – evidence number omitted – each of such countermeasure is respectively examined in the Safety Review, and it is confirmed that the basic design of the Reactor Facilities can assure the safety relating to these countermeasures.

Therefore, first of all, we will examine the specific review, and then judge the claims by the Plaintiffs.

In evaluating the specific review, among the reviews for the Reactor Facilities, the review performed at the time of the permission of the license amendment for the Reactor of Unit No. 2 was the most detailed based on new knowledge; also, the Reactor of Unit No. 2 is only about 150 meters away from the center of the Reactor of Unit No. 1, and since the review for the ground and earthquakes on the Reactor of Unit No. 2 can be applied to the judgement of the safety of the Reactor of Unit No. 1, we will conduct such examination mainly; afterwards, we will judge the validity in comparison with the review criteria of the Reactor of Unit No. 2 regarding the review performed at the time of the permission of the license amendment for the Reactor of Unit No. 1.

According to each exhibit and – evidence number omitted – the specific review for the Reactor Facilities is as follows.

(1) Ground

In the Safety Review, as follows, it is confirmed that the ground of the Reactor Facility Site does not have risks of landslides and mountain tsunamis that may damage the Reactor Facilities and also has the soil bearing capacity required for supporting reactor facilities and no risk of causing ground breakage by earthquakes and differential settlement by loads, and as a result, it is confirmed that the Site has no problem in ensuring the safety as a reactor facility site.

a. Geological survey or the like

Regarding the geological substructure and geological structure around the Site, in addition to the existing topographic map, geological map and literature on geology, it is confirmed that the necessary investigations are conducted through aerial photo interpretation and surface geological survey for land areas and through ultrasonic explorations around the site and others for sea areas. As a result of the evaluations for these survey contents, survey results, and their reliability, all of them are confirmed to be appropriate.

Regarding the geology and ground of the site, it is confirmed that various surveys on the geology and geological structure (surface geological survey, mine survey exploration, trench survey, boring survey or the like) and various tests on rocks and physical properties of bedrock (in-situ rock test in the test mine near the planned core site, indoor rock tests by samples taken from the test mine and boring core, and others) are conducted. As a result of the evaluations for these surveys and tests and their reliability, all of them are confirmed to be appropriate.

b. Evaluation for site ground

It is confirmed that: the ground of the Reactor Facility Site consists of sandstone, shale, and alternating layers of sandstone and shale of the formation of Hagi Beach of Mesozonic Jurassic Oshika group of layers, which are covered with the Quaternary deposits; the ground of the installation planned position of the Reactor Facilities consists of sandstone of Oshika group of layers, shale, and porphyrites which penetrate it; that the strata is inclined 30 to 50 degrees from southeast to south-south east; nine faults are confirmed by the boring surveys, mine survey explorations and others; these slopes are generally high angles. It is confirmed from the interrelationships and characteristics that the faults are old ones that are formed in close relation to the formation of folded structures and that there is no problem regarding activities for safety evaluation.

Additionally, regarding the bearing capacity of the ground, it is confirmed from the result of the flat plate loading tests of the bedrock that it has sufficient bearing capacity against the usual installation pressure and the maximum ground pressure during earthquake, and it is confirmed from the result of stability analysis performed by evaluating the results of rock classification, fault distribution, rock and bedrock testing, and others that the reactor installation ground has sufficient bearing capacity even during earthquake.

Also, regarding slip, it is confirmed from the result of analysis and stability analysis which cover base bottom slip based on the results of block shear tests and others that the reactor installation ground will not cause destruction by slip during earthquake.

Regarding settlement, it is confirmed from the deformation characteristics and analysis result obtained by the flat plate loading tests and others that no settlement with safety trouble will occur.

Furthermore, since there is no landslide that may affect the reactor facilities, steep slopes that may cause landslide or the like, and landslide topography, it is confirmed that the slope collapse that may affect reactor facilities will not occur even during earthquake.

(2) Earthquake

In the Safety Review, as follows, it is evaluated whether earthquakes expected to occur around the reactor site in the future are properly selected from past earthquake history or the like, and, as a result, it is confirmed that such selection is properly conducted.

a. Past damaging earthquakes

Regarding past damaging earthquakes, it is confirmed that the investigations are conducted on earthquakes whose epicenter distance from sites is within 200 kilometers based on the Usami catalogue (1979), Utus catalogue (1982), earthquake catalogue by Japan Meteorological Agency (JMA), and Shiryo Nihon Higai Soran (1983) or the like and that, among these damaging earthquakes, as earthquakes assumed to cause seismic intensity greater than 5 of JMA seismic intensity scale on sites, the earthquake in Sanriku coast (869), earthquake in Rikuzen (1646), earthquake in Rikuchu Morioka (1874), earthquake in Sendai (1835), earthquake in Sanriku offshore (1896), earthquake in Sendai offshore (1897), earthquake in Iwate offshore (1898), earthquake in northern Miyagi (1900), earthquake in Iwate offshore (1905), earthquake in Sanriku offshore (1933), earthquake in Mt. Kinka offshore (1936), earthquake in Mt. Kinka offshore (1937), earthquake in the east coast of Fukushima-prefecture (1938), and earthquake in the Miyagi-prefecture coast in 1978 (1978) are selected.

The selection of these earthquakes and the evaluation of their scale, epicenter distances, and others are carried out by making comparisons and examinations of various sources, and all of them are confirmed to be appropriate.

b. Active faults

It is confirmed that, regarding active faults, based on the investigation on sources and relevant literature such as Nihon no Katsudanso (1980) and Katsukozozu-Akita (1983) or the like, as the faults that may cause earthquakes that affect the site, the west fault of Mt. Jobon, fault of Mt. Kagobo-Mt. Nonodake, fault in sea area, and others are selected, and that locations, scales, activities and others are evaluated. As a result of investigations on the relevant literature to these selected faults and others, investigations related to minute earthquakes, aerial photo interpretation, investigations on the results of ultrasonic exploration, on-site inspections and others, it is confirmed that the cause of the lineament of the west fault of Mt. Jobon is the result of differential erosion due to geological differences or rocky differences, and the cause of the lineament of the fault of Mt. Kagobo-Mt. Nonodake is the differential erosion due to the difference in rocky quality on both ends of the lineament. Regarding the fault in sea areas, as for the four faults in sea areas around the site including the F16 fault (approximately 6.4 kilometers in length), F17 fault (approximately 9.2 kilometers in length), F18 (approximately 6.5 kilometers in length), and F19 fault (approximately 8.9 kilometers in length), it is confirmed from the relationship between the fault and its upper stratum that it is not possible to deny the possibility that the activity is in the late Quaternary. Although the literature shows other faults or lineaments in the and sea area around the site, it is confirmed that the degree of impact by earthquake motions in the site is smaller compared to the fault of such sea area.

c. Seismic geological structure

It is confirmed that: earthquakes that occur farther away from waters close to Miyagi offshore and around Japan Trench do not exceed the impact of the earthquake in Sanriku Coast (869); the upper limit of an earthquake scale that may occur in the crust of Tohoku Japan is expected to be approximately 7.75 magnitude; even if such an upper-scale earthquake is assumed in the fault group that covers from the southwest of Morioka City to the west of Mizusawa City, the impact of this earthquake on the site does not exceed the impact of the earthquake assumed in Miyagi offshore. As for the earthquake assumed from the seismic geological structure, it is confirmed that it is reasonable to assume an earthquake with a magnitude 7.6, the largest scale of earthquakes occurred in waters close to Miyagi offshore, on the plate boundary (20 kilometers of epicentral distance, 45 kilometers of earthquake focal depth) which is considered to have the greatest impact on the site.

(3) Aseismic design of the Reactor of Unit No. 2

In the Safety Review, as follows, it is evaluated whether the design basic earthquake ground motion on the reactor site base is set with sufficient safety allowance in careful consideration to the impact of the earthquake assumed in above-mentioned (2) on the reactor site, and also, regarding the reactor facilities which are subject to the application, evaluated whether it is possible to implement aseismic design with sufficient safety allowance against the set design basic earthquake ground motion from the engineering and technological viewpoints, and, as a result, it is confirmed that, in the basic design of the Reactor of Unit No. 2, such design basic earthquake ground motion is set properly and that the aseismic design is properly implemented.

a. Maximum design basis earthquake and design limit earthquake

It is confirmed that: as earthquakes defined as maximum design basis earthquake, the earthquake in Sanriku coast (in 869, 8.6 magnitude, 201 kilometers of epicentral distance) and the earthquake in Sendai offshore (in 1897, 7.4 magnitude, 48 kilometers of epicentral distance) are selected; as design limit earthquake, the earthquake by the F16 fault (6.2 magnitude, 12.1 kilometers of epicentral distance) and the earthquake by the F16 fault (6.5 magnitude, 21.0 kilometers of epicentral distance) are selected as those assumed from the active fault, and earthquakes assumed on the plate boundary of the waters close to Miyagi offshore (7.6 magnitude, 20 kilometers of epicentral distance, 45 kilometers of earthquake focal depth) is considered as assumed from the seismic geological structure; furthermore, the earthquake directly above its epicenter (6.5 magnitude, 10 kilometers of epicentral distance) is considered. It is confirmed that all of these are appropriate in light of the "Regulatory Guide for Reviewing Seismic Design of Nuclear Power Reactor Facilities" (formulated by the Nuclear Safety Commission (NSC) on July 20, 1986. Hereinafter called the "Regulatory Guide for Reviewing Seismic Design").

b. Basis ground motion

(a) Ground motion characteristics

The maximum amplitude of ground motion, frequency characteristics, time-dependent changes in duration and amplitude envelope are determined by using the empirical formula proposed mainly on the basis of the seismic observation results on the rock and are confirmed to be appropriate.

(b) Basic ground motion

It is confirmed that the basic ground motion is represented by the design response spectrum formulated in consideration of earthquakes being subject to maximum design basis earthquake and design limit earthquake and by the design simulated seismic wave established to suit it, and all of them are confirmed to be appropriate.

c. Classification of importance in aseismic design

It is confirmed that: reactor facilities are classified into three classes, A, B, and C, in accordance with the importance of functions required during an earthquake and are respectively designed to withstand the seismic force required by static or dynamic analysis; among the Class A facilities, the particularly critical facilities are classified as Class AS, and, in this class, the main facilities, equipment and pipe that constitute pressure boundary such as pressure vessel, control rod and control rod driving mechanism, containment vessel, and residual hear removal system, are included. All the classifications of importance for these facilities are confirmed to be appropriate.

d. Seismic force

It is confirmed that: in addition to the static seismic force calculated based on layer shear force coefficient determined in accordance with importance, for Class A facilities, dynamic analysis is performed by using an input earthquake motion obtained from basic ground motion S1, and the seismic force is to be applied as an answering response to it; for Class AS facilities, dynamic analysis is performed by using an input earthquake motion obtained from basic ground motion S2, and the seismic force is to be applied as an answering response to it; the basic ground motion is defined on the free rock surface assumed in the formation of Hagi Beach of Mesozonic Jurassic Oshika group of layers in the site. All the calculations and policies for the application of these seismic forces are, in light of the "Regulatory Guide for Reviewing Seismic Design," confirmed to be appropriate.

e. Combination of loads and allowable limits

As for the combination of loads and allowable limits applied to each class, in light of the "Regulatory Guide for Reviewing Seismic Design," all of them are confirmed to be appropriate.

Also, it is confirmed that, for the equipment required to operate during earthquake, the operational function is not hindered by experiment or the like, and even if there is no fear of being caused by an earthquake, the loads during accident whose effects last long is to be combined with the seismic force by the load and basic ground motion S1 or with the static seismic force.

(4) Aseismic design of the Reactor of Unit No. 2

In the Safety Review, as follows, the Reactor of Unit No. 1 is evaluated in the same aspects, and as a result, it is confirmed that, in the basic design of the Reactor of Unit No. 1, such design basic ground motion is set properly and that the aseismic design is properly implemented.

a. Classification of importance in aseismic design

It is confirmed that: reactor facilities are classified into three classes, A, B, and C, and the aseismic design is implemented in accordance with them; the facilities whose loss of function may cause nuclear accident and the facilities that are critical to prevent disasters in the surrounding public, such as nuclear reactors, nuclear buildings and others, are classified as Class A; among Class A facilities, the facilities particularly critical for safety measures, such as containment vessels (drywell, suppression chamber, vent pipe, and each through hole), control rod, control rod driving mechanism, boric acid injection related facility or the like, are classified as Class AS; the facilities related to high radioactive substances, such as turbine facility and waste treating facility, are classified as Class B, and the other facilities are classified as Class C. The classification of importance for these facilities are confirmed to be appropriate.

b. Aseismic design of each facility

It is confirmed that: reactor facilities are in principle rigid structure, and important buildings are supported directly by the rock; the aseismic design of Class A buildings and structures is implemented by the value which is not less than three times of the horizontal seismic coefficient obtained from dynamic analysis by the seismic wave with a maximum acceleration of 250 gal on the base or horizontal seismic coefficient indicated in the Building Standard Law; the vertical seismic coefficient is constant in the height direction of the buildings and structures and is not less than 1.5 times of the horizontal seismic coefficient indicated in the Building Standard Law; in this case, the seismic forces in horizontal and vertical directions act in the disadvantageous directions at the same time; the aseismic design of Class A equipment and pipes depends on the horizontal seismic coefficient obtained from dynamic analysis by the seismic wave with a maximum acceleration of 250 gal on the base; however, the horizontal seismic coefficient in this case is not less than 1.2 times of the support structures on the installation position; the vertical seismic coefficient is not less than one-half of the horizontal seismic coefficient on the building base part and acts in the disadvantageous directions at the same time with horizontal seismic coefficient; also the displacement and deformation caused by these seismic motions do not hinder function maintenance; for the Class AS facilities, in addition to being treated as Class A, the dynamic analysis by the seismic wave with a maximum acceleration of 375 gal on the base is performed and their functions are maintained; for the Class B and Class C facilities, the aseismic design is respectively implemented by the values of 1.5 times and 1 time of the horizontal seismic coefficient indicated in the Building Standard Law. In the basic design of the Reactor of Unit No. 1, it is confirmed that such design basic earthquake ground motion is set properly, and the aseismic design is properly implemented.

(5) Regarding hydrology

In the Safety Review, it is confirmed that: regarding the tide level in the sea area around the site, according to the observations from 1943 to 1984 by the JMA Ayukawa tide-gage station located approximately 11 kilometers south of the site, the highest tide level is O.P. (It is the base plane for construction of the Nuclear Power Plant, and the mean sea level of Tokyo Bay is -0.74 meters.) +3.22 meters during Chile Earthquake Tsunami (May 24, 1960), and the lowest tide level is O.P. -2.96 meters during the same tsunami; regarding high waves, although the maximum wave height is 6.83 meters according to the observations approximately 500 meters off the coast from April 1981 to March 1986, waves are shielded by the breakwater installed in front of the site; regarding the rise of water level due to tsunami, as a result of the examinations of the documentary searches on the past tsunamis and others, even in consideration of the synodic mean high tide, it is approximately O.P. +9.1 meters at maximum, whereas the main facilities, such as the Nuclear Reactor Buildings and others, are installed on a site of more than O.P.+14.8 meters. As a result, it is confirmed that the rise of water level due to high waves and tsunamis does not hinder the safety of the Reactor Facilities.

(6) Validity of examining methods for safety of Reactor of Unit No. 1

According to the above-confirmed specific review from (1) to (5), although it can be confirmed that the safety judgement related to the natural locational conditions of the Reactor Facilities in the Safety Review is conducted based on the reasonable ground, the review methods for the seismic designs are different between the Reactor of Unit No. 1 and the Reactor of Unit No. 2, and since the examination on the Reactor of Unit No. 2 reflects the latest findings, we will further examine the validity of the review method for the Reactor of Unit No. 1 in light of the review method for the Reactor of Unit No. 2.

a. Regulatory guide for reviewing seismic design

According to – evidence number omitted – the NSC established the Regulatory Guide for Reviewing Seismic Design on July 20, 1981, and, according to this, the review for seismic design finds that: the reactor facility for power generation must have sufficient seismic resistance against any anticipated seismic force so that this does not cause a major accident, and the basic policy is that buildings and structures must be rigid in principle and that critical buildings and structures must be supported by bedrock; in the specific review, the importance of each facility in the seismic design of nuclear reactor is, in terms of the environmental impact of radiation that may occur due to earthquake, classified into Class A (facilities that contain radioactive substances by themselves or that are directly related to the facilities that contain them and may release radioactive substances into the outside due to their loss of function, facilities that are necessary to prevent these situations, and facilities that are necessary to reduce the impacts of discharged radioactive substances during these accidents and have large influences and effects. Among Class A facilities, particularly important facilities are categorized into Class AS.), Class B (facilities that have relatively small influences and effects in such situation), and Class C (facilities that are neither A nor B and are required to maintain safety equivalent to that of general industrial facilities); each Class A facility bears the greater seismic force of either the seismic force of maximum design basis earthquake (the largest-impact earthquake assumed from earthquakes that may have the same impact, if occur again, on the site and its neighbor as that of the earthquakes that are considered from historical documents to have had the impacts on the site and its neighbor in the past and from earthquakes by the high active faults that may have the impact on the site in the near future) or the static seismic force; each Class AS facility can maintain their safety functions against the seismic force of design limit earthquake (the largest-impact earthquake assumed, based on seismological standpoints, by examining the earthquakes that exceed the maximum design basis earthquake from engineering standpoints based on the occurrence state of past earthquakes, the active faults around the site, and the structure of earthquake zone); each Class B facility bears the static seismic force, and, as for the facilities that have the possibility of resonance, such influence is examined; each Class C facility can bear the seismic force; furthermore, the belongings to the higher classification do not suffer rippled damage due to the damage of the belongings to the lower classification.

b. Regarding classification of importance in aseismic design

Comparing the classification of importance in aseismic design in above-confirmed Regulatory Guide for Reviewing Seismic Design with the classification of importance in aseismic design in the safety review for the Reactor of Unit No. 1, although there are some differences confirmed in detail, it is confirmed that they are substantially the same as a whole, and according to such facts, even in light of the Regulatory Guide for Reviewing Seismic Design in this regard, it can be said that the safety review for the Reactor of Unit No. 1 still has validity.

c. Regarding design basis ground motion

As previously found, in the aseismic design of the Reactor of Unit No. 1, the Defendant set the design maximum acceleration on the base rock as 250 gal and as 375 gal regarding particularly important Class AS facilities for safety, and this point was confirmed to be appropriate in the Safety Review.

In the examinations of the ground for setting such design maximum acceleration, it is confirmed that: according to – evidence number omitted – and the testimony by a witness, Seiken Ogata, the earthquakes that had their epicenters in and around Miyagi-prefecture and caused damages in the same areas can be classified into three earthquake groups based on the epicenter distribution, (i) the earthquake in the waters close to Miyagi-prefecture (earthquake with an epicenter of approximately 50 to 100 kilometers east of Miyagi-prefecture), (ii) the earthquake in the waters away from Sanriku offshore (earthquake with an epicenter near the Japan Trench which is approximately 200 kilometers from the Sanriku coast), and (iii) the earthquake in the inland of Miyagi-prefecture (earthquake with an epicenter around the northern part of Miyagi-prefecture between the Kitami Mountains and the Ou Mountains); by selecting the earthquakes that are considered to have the largest impacts on the Nuclear Power Plant Site from each of such three earthquake groups and assuming the maximum acceleration of the seismic motion of such earthquakes on the base rock of the Reactor Buildings by using the Kanai formula (experimental formula that expresses the relationship between the maximum acceleration of ground motion on bedrock, epicentral distance, and magnitude made by Kiyoshi Kanai, Professor, Earthquake Research Institute, The University of Tokyo), the earthquake in Sendai offshore (with 7.8 magnitude) on February 20, 1897, resulted in 184 gal for (i), the earthquake in Sanriku coast (with 8.6 magnitude) on July 13, 869, resulted in 104 gal for (ii), and the earthquake in northern Miyagi (with 7.3 magnitude) on May 12, 1900, resulted in 146 gal for (iii); on the other hand, according to the expected value chart of the maximum acceleration of seismic motion that may happen once in 200 years provided by the Kawasumi map (distribution map of earthquake risk for all of Japan created by Hiroshi Kawasumi, Professor of Earthquake Research Institute, University of Tokyo, which obtains the ground motions expected once in 75 years, 100 years and 200 years at the 345 mesh points collected by dividing all of Japan in every 0.5 degrees at both longitude and latitude and shows such ground motion as the shape of contours by referring to the epicenter positions and magnitudes of 343 earthquakes that occurred in and around Japan from 599 to 1949), the value of the nearest isoline on the Nuclear Power Plant Site is 500 gal (the maximum acceleration on the ground surface); by deducting the amplification factor (2.78) between the ground surface and the foundation rock obtained as a result of the defendant’s seismic observation on the site from such figure, the maximum acceleration of the ground motion that should be expected to occur on the foundation rock of the Reactor Building in the future is calculated to be 180 gal; also, according to the expected value of the foundation maximum speed by Kiyoshi Kanai and the distribution map of foundation speed by Yukio Otsuki, which show the maximum speed of ground motion on the bedrock during earthquake, as for the areas including the Nuclear Power Plant site, the value of 8 centimeters per second is the most severe value for assuming the maximum acceleration of ground motion, and, the maximum acceleration of the ground motion that should be considered to occur on the foundation rock of the Reactor building in the future is calculated to be 168 gal by such value; as above, since the largest maximum acceleration estimated from the past earthquakes is 184 gal and the largest maximum acceleration estimated from the studies that statistically expressed the expected value of the intensity of the ground motion is 180 gal, the Defendant sets a maximum acceleration for design on the foundation rock of the building of the Reactor of Unit No. 1 to 250 gal with a margin, and, for Class AS facilities such as containment vessel, control rod driving mechanism and others that are particularly critical for safety, to 375 gal.

Also, as previously stated, the seismic design of the Reactor of Unit No. 2 is carried out in accordance with the Regulatory Guide for Reviewing Seismic Design, and as for the foundation bedrock of the building of the Reactor of Unit No. 2, the basic ground motion is set by considering the earthquake in Sendai offshore in 1897 and the earthquake in Sanriku coast in 869 are as maximum design basis earthquake whereas considering the earthquake near the F16 and F17 faults in the sea area around the Nuclear Power Plant and the plate boundary (waters close to Miyagi offshore) and the earthquake directly above its epicenter as design limit earthquake, and according to – evidence number omitted – and the testimony by a witness, Seiken Ogata, looking at the maximum acceleration from the waveform of simulated seismic wave for design of such basis ground motion, it is confirmed that maximum design basis earthquake is approximately 250 gal whereas design limit earthquake is approximately 375 gal.

Therefore, when the Reactor Facilities of Unit No. 1 was designed, since there was no aseismic design guideline for nuclear power reactor facilities, the aseismic design did not follow such guideline, and the review of the aseismic design was not conducted along with the formulation of such guideline. Additionally, even if the earthquake near the F16 and F17 faults in the sea area around the Nuclear Power Plant and the plate boundary (waters close to Miyagi offshore) and the earthquake directly above its epicenter are not taken into account in setting the maximum acceleration of the Reactor of Unit No. 1, considering in the same way as the Reactor Facilities of Unit No. 2 that the Reactor Facilities of Unit No. 1 carried out the aseismic design by assuming the earthquake in Sendai offshore in 1897 and the earthquake in Sanriku coast in 869 as the past earthquakes and that the assumed maximum accelerations are approximately 375 gal for the Class AS facilities and approximately 250 gal for other facilities respectively, even in light of the Regulatory Guide for Reviewing Seismic

Design about setting basis ground motion, it can be said that the safety review of the Reactor of Unit No. 1 still has validity.

(7) Conclusion

According to each fact found from (1) to (6) above, it can be confirmed that the safety judgement in the Safety Review regarding the natural location conditions of the Reactor Facilities was conducted based on the reasonable ground, and considering such facts and above confirmed organization and characteristics of the NSC, it can be assumed that the Reactor Facilities can assure the safety of the natural location conditions unless the Plaintiffs further assert and prove this point.

X. Conclusion

1. Based on the finding above, the holdings of the Plaintiffs' claims based on the personal or environmental rights are summarized as follows.

(1) Firstly, as for the impacts of radiation exposure on human life and body, the existence of a threshold dose is almost clear for acute disorder whereas the existence of a threshold dose has not been fully elucidated as for tardive and genetic disorder, but, as a matter of legal evaluation, it is reasonable to assume that there should be no threshold regarding the relationship between the exposure dose in low dose region and the occurrence of tardive disorder or the like.

(2) On the other hand, although the measures to prevent the release of radioactive substances into general environment are taken in the Reactor Facilities, since it is unavoidable to release a certain amount of radioactive substances into environment by its operation and it is reasonable to assume that there should be no threshold as for the occurrence of disorder by radiation exposure, as were possibility, it cannot be denied that there is a probability of the occurrence of damages on lives and bodies of the Plaintiffs. However, considering the necessity of the Nuclear Power Plant in terms of power supply and demand, the safety required of nuclear reactor facilities, if the possibility of damages on human life and body by radiation is low enough to be ignored under the social standards by taking the countermeasures that reduce the release of radioactive substances as low as possible to prevent the materialization of potential risk and that should keep the risk of accidents by this and exposure dose during usual operation low enough to be ignored under the social standards in any case, should be construed that it is not possible to confirm the claim for injunction based on illegal infringement of the personal or environmental rights as it cannot be said that there is the risk of damages on life and body by operating reactor facilities.

(3) In our country, the effective dose equivalent is defined as 0.1 rem per year for the dose equivalent limit outside environmental monitoring areas at nuclear power plants. The number is determined by respecting the recommendations of the ICRP and receiving the report from the Radiation Review Council, and considering the details of the ICRP recommendations and others, provided that the measures that reduce the public exposure dose as low as possible are taken, it is reasonable to set the radiation dose, which is considered to be so low that the possibility of damage caused by radiation can be ignored under the social standards, to 0.1 rem per year as an effective dose equivalent.

(4) In evaluating the countermeasures related to the basic design of the Reactor Facilities, in the Safety Review, it is confirmed that the basic design of the Reactor Facilities can assure, firstly, the safety related to countermeasures for accident prevention of reactor facilities consisted of the preventive measures for occurrence of unusual events, for expansion of unusual events, and for unusual release of radioactive substances, secondly, the safety related to countermeasures for reduction in exposure during usual operation of reactor facilities, and thirdly the safety related to countermeasures for location conditions of reactor facilities consisted of countermeasures for natural location conditions and for separation of reactor facilities and public, and judging from the specific examinations, it is construed that all decisions by the Nuclear Safety Commission has validity. In addition, the public exposure dose during usual operation of the Reactor Facilities is well below 0.1 rem per year even under the severe conditions, and it is confirmed that it can be kept even lower since the measures that reduce the public exposure dose as low as possible are taken. Therefore, it can be confirmed that the Reactor Facilities has no lack for the countermeasures related to its basic design, and there is no sufficient evidence to reverse this holding.

Also, judging from the countermeasures at the construction and operation stages of the Reactor Facilities, the lacks are not specifically identified.

According to such facts, it can be said that the Reactor Facilities take the sufficient countermeasures to make the possibility of the occurrence of disorders by radiation low enough to be ignored under the social standards.

(5) Furthermore, even considering the reported findings of accidents at the other nuclear power plants, it cannot be said that similar accidents may occur at the Nuclear Power Plant, and they do not overrun such judgement. Also, even considering the events that occurred at the Reactor of Unit No. 1, it is not enough to reverse the above holding.

(6) As above, we cannot confirm that there is the possibility of the occurrence of disorders against the Plaintiffs' lives and bodies by the radiation which is beyond the extent that can be ignored under the social standards by usual operation of the Nuclear Power Plant; also, we cannot confirm that there may be an accident that causes damages to the Plaintiffs' lives and bodies by the radiation which is beyond the extent that can be ignored under the social standards at the Nuclear Power Plant.

2. As stated above, the claim by the Plaintiffs is groundless and shall, therefore, be dismissed, and, regarding the burden of court confirms, under the Articles 89 and 93 of the Law on Civil Procedure shall apply, and the judgement is rendered as described in the main text.