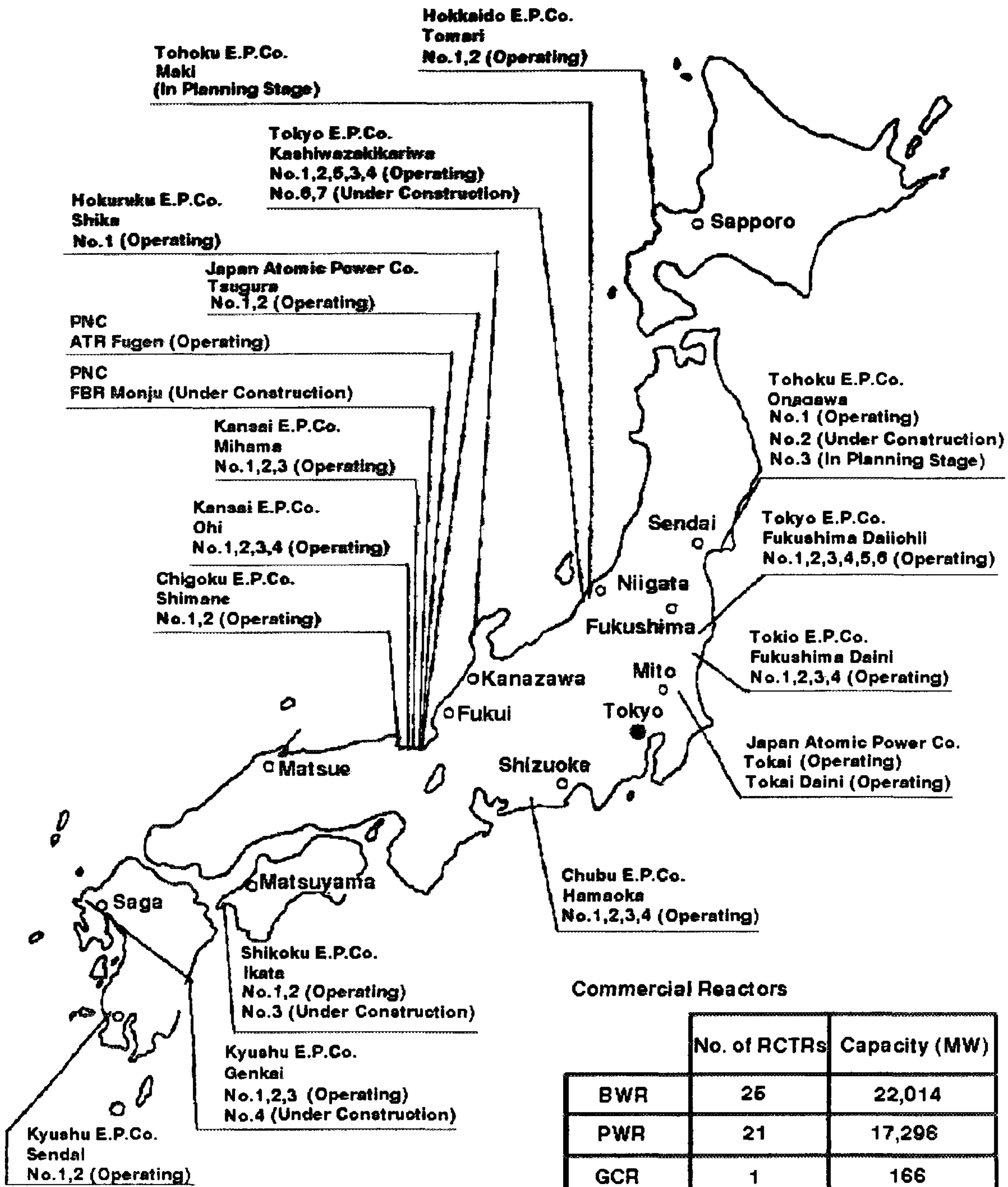
Nuclear Regulatory Guides for LWR (PWR) Fuel in Japan and Some Related Safety Research

Michio Ichikawa

Japan Atomic Energy Research Institute, Tokai-Mura, Ibaraki-Ken, Japan

1 Introduction

More than 25 years have passed since the first nuclear power plant started operation in Japan. At present 47 commercial nuclear power plants (NPPs) are in operation. Their total generating capacity is 39.5 *GWe* which represents approximately 20% of the total generating capacity in Japan. In addition 5 commercial nuclear power plants including two ABWRs are under construction. The sites of these NPPs are shown in Fig. 1.



Commercial Reactors

	No. of RCTRs	Capacity (MW)
BWR	25	22,014
PWR	21	17,296
GCR	1	166
Total	47	39,476

Figure 1 Sites of nuclear power stations in Japan

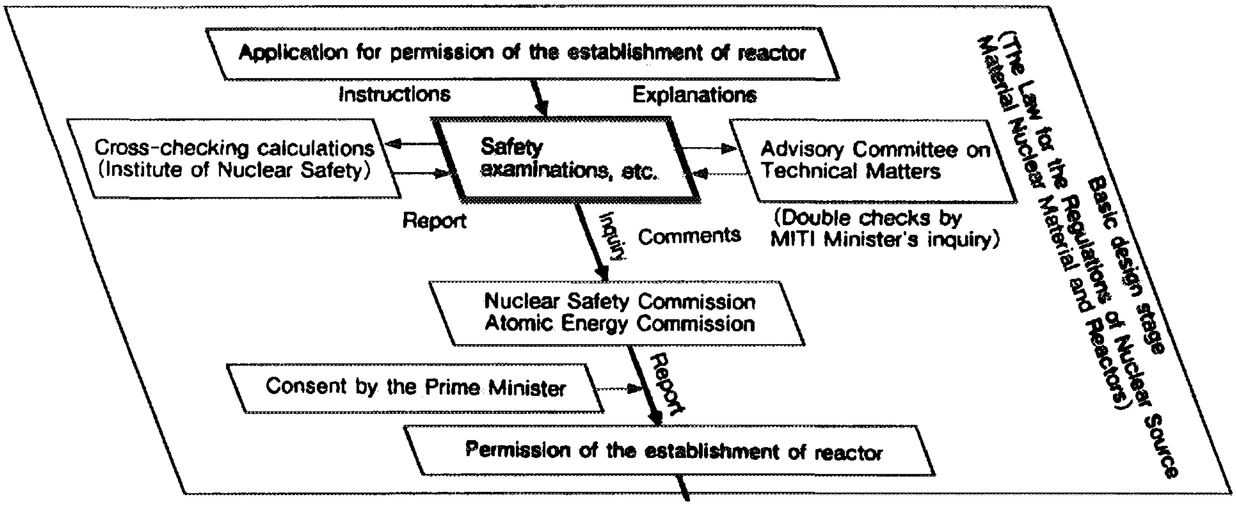


Figure 2 Flow of nuclear power plant's examination

Based upon the recognition of "No nuclear energy in use without assuring nuclear safety", the regulatory authorities enforce strict safety regulations in each stage of design, construction and operation of the NPPs.

In the present paper it is intended to introduce the general licensing procedure in Japan and regulatory guides and expert committee reports closely related to PWR fuel, together with some major results of rector safety research experiments at NSPR (Nuclear Safety Research Reactor of JAERI), which have been used for establishment of related guide.

Because of the limited space of the paper it is not possible to describe these guides and reports in detail. It, however, is intended to describe in some detail on the item which may be of interest to the fuel people in the meeting.

2 General Situation of Licensing of NPP and Regulatory Guides in Japan

2.1 Regulatory Processes for NPP

All the processes of the siting, design, construction and operation of NPP are subject to the control of competent government agencies. The Ministry of International Trade and Industry (MITI) is responsible for the regulation of commercial NPP, and the Science and Technology Agency (STA) is responsible for the regulation of research reactorsand of reactors under development. On the other hand, as an advisory organization for the prime minister, the Nuclear Safety Commission (NSC) reviews the regulatory processes and administrative processes conducted by the competent agencies (MITI, STA) from safety point of view and make necessary recommendations to the government.

The major licensing processes of the commercial NPP are divided into three stages; application for installation permit, application for construction permit and inspection during construction and operation phases. In the present paper, only the first stages are described generally as follows.

In the application for installation permit for commercial NPP an applicant who intends to install (or modify) NPP must submit to MITI the documents that includes the description of the site conditions, basic plant design and relevant safety evaluation. The MITI conducts the first review of the application with the aid of Technical Advisory Committee on Nuclear Power, MITI.

After the approval by MITI the NSC independently conducts the second review with the aid of Committee on Examination of Reactor Safety, NSC. The conclusion of NSC's review is reported to MITI. If the applicant is eventually approved by NSC, the MITI issues the license indicating the acceptance of the basic design of the plant for the proposed site.

The general correlation of the regulatory processes as described above is shown in Fig.2.

2.2 Major Regulatory Guides and Related Committee Reports on LWR Fuel [1]

The major regulatory guides which are closely related to LWR fuel are as follows:

(1) Guide for Safety Design of Light Water Nuclear Power Reactor Facilities (1990)

(2) Guide for Safety Evaluation of Light Water Nuclear Power Reactor Facilities (1990)

(3) Guide for Evaluation of Emergency Core Cooling System Performance in Light Water Nuclear power Reactors (1981, Rev. 1988)

(4) Guide for Evaluation of Reactivity Initiated Events in Light Water Nuclear Power Reactor Facilities (1984)

In the Guide (1), the major item which is related with fuel is as follows (14. Fuel design):

(i) The fuel assemblies shall be designed not to loose their integrity despite various unfavorable factors that may take place during their use in the reactor.

(ii) The fuel assembly shall be designed not to be excessively deformed during transport or handling.

In the Guide (2), the major items which are related with fuel are as follows:

(i) The minimum heat flux ratio or the minimum critical power ratio shall be lager than the acceptable limit.

(ii) Fuel cladding shall not be mechanically damaged.

(iii) Fuel enthalpy shall not exceed acceptable limit.

(iv) Pressure on the reactor coolant pressure boundary shall not exceed 100% of the maximum allowable working pressure.

For the criteria on "Accident":

(I) The core shall not be damaged considerably and adequate coolable state of the core shall be maintained.

(II) Fuel enthalpy shall not exceed the specified limit.

(III) Pressure on the reactor coolant pressure boundary shall not exceed 120% of the maximum allowable working pressure.

(IV) and (V) are skipped as they are not much related with fuel.

In the Guide (3), the major items which are related with fuel are as follows:

(i) The calculated maximum fuel cladding temperature shall nor exceed 1.200°C.

(ii) The calculated stoichiometric amount of oxidation of the fuel cladding shall not exceed 15% of the cladding thickness before significant oxidation.

The general content of Guide (4) is described in the next chapter.

There are several expert committee reports which are resolved or approved by NSC. The reports which are closely related with licensing of the LWR fuel are as follows:

(1) On the 17×17 Fuel Assemblies used in PWR (1976)

(2) On the Fuel Design Methodology of LWR (1988)

(3) Clarification of "Fuel Cladding Should not Fail Mechanically" (1985)

The general content of these reports is described in the chapters 4 - 5.

3 Regulatory Guide "Evaluation Guide for Reactivity Initiated Events in Light Water Power Reactor" (1984)

3.1 Objective

This guide aims to confirm the integrity of reactor core and reactor pressure boundary by evaluating fuel enthalpy increase due to rapid insertion of positive reactivity which results in reactor power increase in the reactor core under or near the critical conditions.

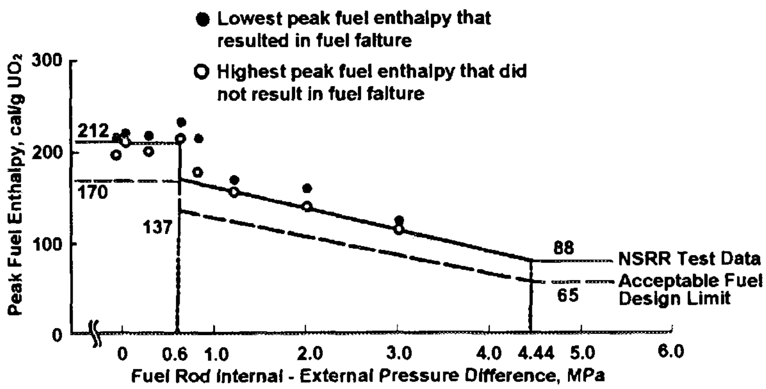


Figure 3 Fuel failure thresholds as a function of fuel rod internal-external pressure difference

3.2 Position and Range for Applications

This guide is to evaluate the reactivity initiated events of PWR and BWR following the regulatory guides "Guide for safety design of light water nuclear power reactor facilities" and "Guide for safety evaluation of light water nuclear power reactor facilities". The present guide shall be reviewed when knowledge like design improvements, accumulation of experience, etc. are obtained.

The basic idea of the present guide may be applied to the evaluation of reactivity initiated events of the light water reactors or heavy water reactors which have similar fuel structure and compositions.

3.3 Judgment Criteria

3.3.1 <u>During anticipated operational transient</u>

(i) The maximum fuel enthalpy shall not exceed which are decided depending on the differential pressure between fuel internal pressure and coolant pressure as shown in Fig. 3 (dotted line).

(ii) The pressure on the reactor coolant pressure boundaries shall be 1.1 times maximum working pressure and below.

3.3.2 <u>During the accident</u>

(i) Maximum fuel enthalpy shall not exceed 230 *cal/g* of UO_2.

(ii) The pressure on the reactor coolant pressure boundaries shall be 1.2 times maximum working pressure and below

(iii) During anticipated operational transition and accident, shock pressure from the burst of water logged fuel shall not result in any damage to the reactor scram ability and integrity of reactor pressure vessel.

3.4 Requirements for the Analyses

The initial condition of reactor states, dynamic characteristics calculation, fuel behavior analyses, pressure surge calculations and mechanical energy generation by the burst of waterlogged fuel are described. The details are not given here.

3.5 Required Documents for Evaluation

Several documents including calculation code, major input and sensitivity analyses of the calculation programs are required. The details are omitted here.

4 Report by the Committee on Examination of Reactor Safety "On the 17×17 Fuel Assemblies Use in PWR" (1976)

4.1 Introduction

The purpose of this report is to investigate generally the basic design of Westinghouse 17×17 fuel assemblies to be used in PWR. The investigation includes structural, nuclear and thermohydraulic design of the assemblies. Considering the interest in the present meeting the emphasis on the description of the report is placed on the structural design.

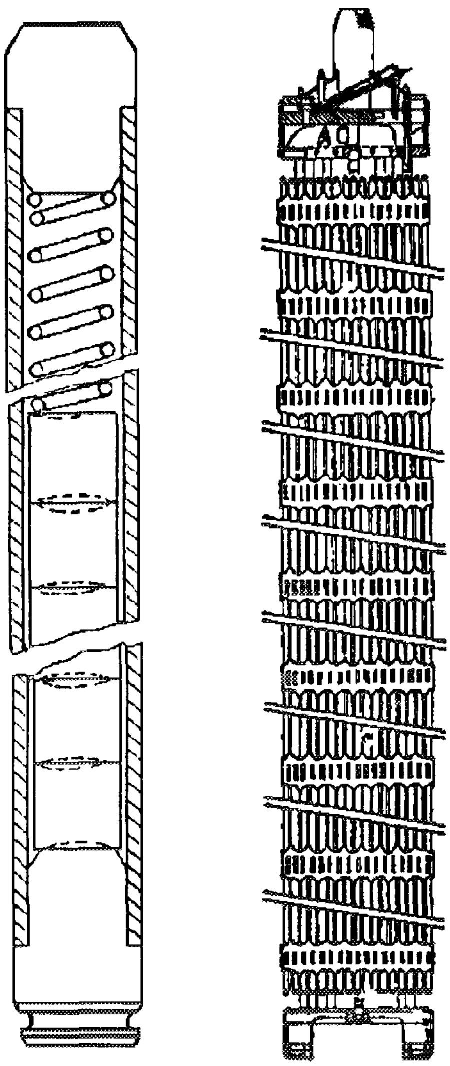


Figure 4 General design of PWR fuel rod and assembly (17×17 type)

4.2 Design of the 17 ×17 Fuel Assemblies

4.2.1 General

The design of the fuel rod is shown in Fig. 4 and the fuel rods are held by grids. Fuel assembly is composed of these rods and a skeleton which is formed by 24 control guide thimbles, 9 grids and upper/lower nozzles. The general design of 17 ×17 fuel assembly is show also in Fig. 4.

4.4.2 Design criteria

To assure the integrity of the fuel during normal operation conditions and anticipated operational transients, three criteria for structural design, nuclear design and thermohydrauliv design are established.

(a) Structural design criteria

Structural design criteria are designed as follows:

(i) Maximum fuel temperature shall be below melting point of uranium dioxide.
(ii) Inner pressure of fuel rod shall be within the operating coolant pressure (157 kg/cm^2g). This criteria has been changed in 1988 to allow the internal pressure to exceed the outer pressure under certain conditions (described in 5.4.1)
(iii) Stress in the cladding shall be within the proof stress of the Zircaloy-4.
(iv) The change of the circumferential tensile strain of the cladding during each transient shall not exceed 1%.
(v) The cumulative number of fatigue cycles shall not exceed the design fatigue life limit.

In addition to the above statements, fuel failure due to cladding fretting corrosion, cladding flattening, etc., shall be prevented.

Concerning the control guide thimbles, grids, upper/lower nozzles and the places close to these parts, the following criteria are established to ensure the integrity and avoid the influence of the surrounding structures:

(i) In principle the stress due to static and cyclic load in the normal operation and anticipated operational transient shall be evaluated based on ASME Section III.
(ii) Concerning the load during transportation and handling, significant deformation shall not take place for 6G (G denotes gravity acceleration).

(b) Nuclear design criteria

Six criteria are described, including peaking factors, maximum fuel temperature (below melting point of uranium dioxide), Doppler coefficient, maximum reactivity insertion. The details are not given here.

(c) Thermohydraulic design criteria

Four criteria are described including minimum DNB ratio (above 1.30), maximum fuel temperature (as mentioned above), coolant flow (above 95.5% of design value) and prevention of thermohydraulic instability. The details are not given here.

4.3 Result of Investigation

4.3.1 Structural Design

(a) Fuel design code

The code performs analyses for full length of fuel rod in 2D (R, Z). The code contains various submodels concerning fuel behavior and can calculate fuel temperature, pellet/cladding gap size, fission gas release, rod internal pressure, cladding deformation, cladding stress etc. as a function of burnup. The code is considered appropriate for the use of design of 17×17 fuel, judging from the comparison with the measurements and large amount of records of performances of the fuel designed and manufactured based upon the code.

(b) Fuel internal pressure (criteria has been changed)

(c) Cladding stress

For the evaluation of the cladding stresses, equivalent stress of the volume average are calculated and compared with proof stress of Zircaloy-4. The cladding stresses thus evaluated have been confirmed that they do not exceed the proof stress of cladding during normal operation and anticipated operational transients and have sufficient margins.

(d) Cladding strain

The 1% strain criterion is based on the irradiation performance records and judged appropriate. The evaluation of the cladding average circumferential strain during anticipated operational transient confirms that the calculated strain is sufficiently satisfy the 1% criterion The cladding strain due to swelling for long term operation is small enough and will not cause any problem at the end of fuel life.

(e) Cladding cumulative fatigue

The cumulative fatigue of the cladding has been investigated for the failure possibility using Langer-O'Donnel correlation. Integrity of fuel is judged sufficiently assured against the power changes expected during the fuel life time.

(f) Stability of shape and sizes

The results of the investigation on the stability during transportation, rod growth, interaction with rods and nozzles, strength tests of assemblies, assurance of flow are confirmed that the stability of shape and sizes are maintained during the operation.

(g) Fretting corrosion

Fretting corrosion of 17×17 type rods were compared with the existing 14×14 and 15×15 type fuel rods and it was confirmed that the fretting corrosion of 17×17 rods is slightly less than the existing one.

(h) Fuel rod bowing

As the countermeasure to the fuel rod bowing of Westinghouse design fuel, the present design applies shortening of grid spans and reduction of restraint forces of the grids. The contact between rods is considered to be practically avoided by the new design.

(i) Behavior of 17×17 fuel under LOCA conditions

Westinghouse has simulated the behavior of fuel cladding during LOCA by a series of a single rod burst tests. behavior of cladding during simulated LOCA id characterized by swelling and burst temperature. Statistic comparison of the burst behavior of 17×17 type fuel cladding with that of 15×15 shows that both behave similarly.

(j) Others

Concerning densification and flattening of Westinghouse type fuel and general hydriding failure, the countermeasures have been taken and they are almost solved presently.

4.3.2 Nuclear Design

Nuclear analysis codes, linear heat rating of fuel rod, reactivity coefficients and constant axial offset control operation are described. The details are not given here.

4.3.3 Thermohydraulic Design

Thermohydraulic design codes, minimum DNB ratio, maximum fuel temperature and thermohydraulic stability are described. The details are not given here.

4.4 Others

Records of the Performance and Quality Controls been described in a separate chapter. Data, however, are rather old. Therefore they are not described here.

4.5 Conclusion

PWR new type fuel committee has conducted investigation on the 17×17 type fuel and concluded that the use of this fuel in PWR will not result any problem.

5 On the Fuel Design Methodology of LWR (1988)

5.1 Introduction

This report describes the results of investigation on the appropriateness of the revision of the current methodology for extension to the high burnup use of fuel assemblies.

5.2 Background

The current fuel design methodology has been described in the Committee on Examination of Reactor Safety Report "on the 17×17 fuel assemblies for PWR" and has a long term positive achievements. However, as the methodology is based upon conservative criteria and evaluation methods, there could be cases where merits of the design improvements are not properly evaluated.

On the other hand, evaluation codes can predict thermal/mechanical behavior of fuel rods with better accuracy at high burnup region (above 39 000 *MWd/t* for PWR) reflecting recent knowledge. And a new criterion on the internal pressure of the fuel rod has been proposed. Considering this background and based upon investigations on the recently developed/revised codes, the criterion has been examined.

5.3 Results of Investigation

Concerning the internal pressure criterion, the current criterion "fuel rod internal pressure shall be within the operating coolant pressure (157 kg/cm^2g) has been changed". The new criterion is "fuel rod internal pressure shall not exceed the pressure which results the increase of pellet/cladding gap due to outward creep deformation of the cladding under normal conditions".

The acceptance of this criterion is decided appropriate by the following reasons:

(i) Fuel rod internal pressure is restricted by the cladding stress criterion against mechanical failure. From the stand point of prevention of excessive increase of fuel temperature, generation of thermal feedback will be prevented by the new criterion.

(ii) The fuel design code to be used for evaluation is able to predict fuel rod behavior under irradiation and uncertainties of the prediction are well evaluated quantitatively by the fuel irradiation and other data.

It is expected that the review will be conducted according to the new information.

5.4 Appendix. PWR Fuel Design Methodology

5.4.1 Fuel Rod Internal Pressure Criterion

The revised PWR design criterion has adopted new fuel internal pressure criterion considering the conservatism of the current criterion in comparison with the cladding stress criterion.

(1) Prevention of Thermal Feedback

During normal operating conditions the fuel internal pressure is below the operating coolant pressure (outer pressure). Cladding diameter decreases due to the inward creep deformation and the cladding comes to contact with pellets. Hereafter internal pressure increases due to fission gas release and accumulation and the internal pressure may exceed the outer pressure at high burnup region. In this condition the once closed pellet/cladding gap may open again due to the outward creep deformation of the cladding, which results the increase

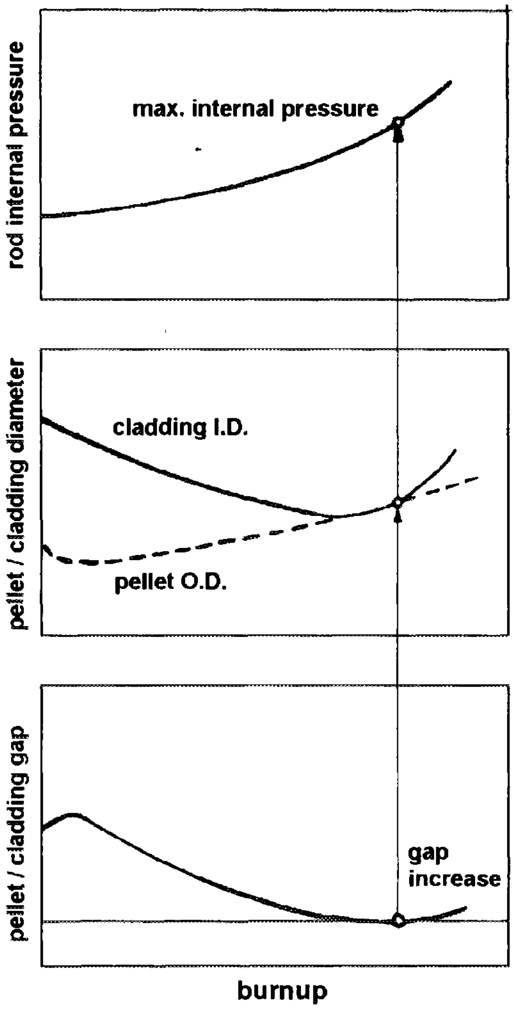


Figure 5 Method to determine the maximum internal pressure [1]

of cladding diameter. Gap conductance may decrease and fuel temperature may rise due to the opening of the gap. This may result further release of fission gas and then internal pressure continues to rise. This may result the further opening of the gap. This phenomena is called thermal feedback and the use of the fuel may result the excess increase of fuel temperature. The new criterion for fuel internal pressure is established to avoid the thermal feedback.

(2) Method to Decide the Internal Pressure Standard

(a) Analysis conditions
Fuel rod power histories are chosen among power histories to be used in the practice to give the severer fuel internal pressures.

(b) The maximum internal pressure which does not result gap increase (Fig. 5)
From the calculation of pellet/cladding gap change by the fuel rod design code the time of gap increase (or reopening of once closed gap) is determined. The fuel rod internal pressure of this point is the maximum internal pressure.

(c) Fuel rod internal pressure standard
By the method described in (b) the lower envelope of the maximum internal pressure for each

power histories is determined. In addition the uncertainties of the analysis (uncertainties arising from design code and fabrication parameters) are taken into consideration and herewith determined value is called as the fuel rod internal pressure standard.

5.4.2. Fuel Design Code

(1) Improvement of the design code

PWR fuel rod design codes have been improved by the high burnup data covering the PWR operating conditions to obtain better accuracy.

Concerning the revised Mitsubishi fuel design code, fission gas release model has been changed to be more burnup dependent by applying the model in which the fission gas release becomes significant after the fuel burnup exceeds given threshold at high burnup region. Also creep model of cladding and swelling/densification model of pellet have been improved.

Concerning revised Nuclear Fuel Industries fuel design code, diffusion constants of the fission gas release has been revised and pellet relocation model has been used.

(2) Verification of the improved fuel design codes

The verification of the codes are based upon wide range of data, including data from Halden reactor, post irradiation examination of power reactor fuel, international cooperation such as Overramp project and others. By these verification activities it has been confirmed that the calculations by improved codes give better agreement with the measurements. In the evaluation of the design codes it is appropriate to consider that the differences between calculation and measurements are taken as the uncertainties of the codes.

6 Clarification of the Statement "Fuel Cladding Should Not Be Mechanically Damaged" (Expert Committee Report, 1985)

In the Safety Evaluation Guide it is required that "fuel cladding shall not be mechanically damaged" in the case of anticipated operational transients. In the safety evaluation of LWR fuel design, the above statements are embodied by the following criteria:

- the circumferential average strain of fuel cladding shall be less than 1% (equivalent of 180% of the surface heat flux) (BWR);
- the maximum fuel centerline temperature shall be less than the melting point of uranium dioxide.

Therefore in the safety evaluation it is confirmed that the surface heat flux (BWR) or centerline temperature (PWR) fulfill these criteria. In this committee report it is investigated that the appropriateness of judging of mechanical damage by the thermal indexes and also the requirement of the expression "fuel cladding shall not be mechanically damaged" is clarified.

6.1 The Requirement of "Fuel Cladding Shall Not Be Mechanically Damaged"

This statement means that "penetrating damage shall not systematically be generated due to mechanical loading on the claddings".

6.2 Embodied Criteria (PWR)

(i) During the anticipated operational transients of the current reactor types and fuels, damage modes which may cause penetrating damage to the cladding are discussed. To prevent the systematic damage during anticipated operational transient, it is concluded that the corresponding criteria to "fuel cladding shall not be mechanically damaged" can be represented by the condition to prevent overstrain failure of the cladding, i.e. "the circumferential averaged strain of fuel cladding shall be less than 1%". In the case of PWR during the anticipated operational transient, "the maximum fuel centerline temperature shall be less than melting point of uranium dioxide". has been confirmed by the dynamic analyses. By satisfying this condition, the circumferential average strain of fuel cladding is less than 1%. Therefore it is concluded that this evaluation methodology is appropriate.

(ii) Recently the cause of fuel failure during the power ramping tests is identified as stress corrosion of cladding of Zirconium alloys. It is well known that Zircalloy cladding may fail depending on the power histories even with strain less than 1% or with centerline temperature below melting point. The investigation has been conducted concerning this point and the conclusion is that the possibility of systematic fuel failure during the anticipated operational transients for the current reactor type and fuel is considered sufficiently small. Therefore it is concluded that there in no need to change the current fuel failure criteria.

7 Studies of Fuel Behavior During Simulated Reactivity Initiated Accidents (RIA) in NSRR ([2], [3])

7.1 Introduction

RIA research program in Japan was imitated in 1972 as the NSRR (Nuclear Safety Research reactor) program. The results of over 800 experiments with fresh fuel rods have been reflected on the establishment of Japanese regulatory guide "Evaluation Guide for Reactivity Initiated Events in Light Water Power Reactor", issued by Nuclear Safety Commission in 1984. Considering the tendency of high burnup use of LWR fuel, NSRR was modified to be able to handle the preirradiated fuel in 1988. At the same time the operation modes of NSRR were also modified so that various power histories can be given to fuel specimen.

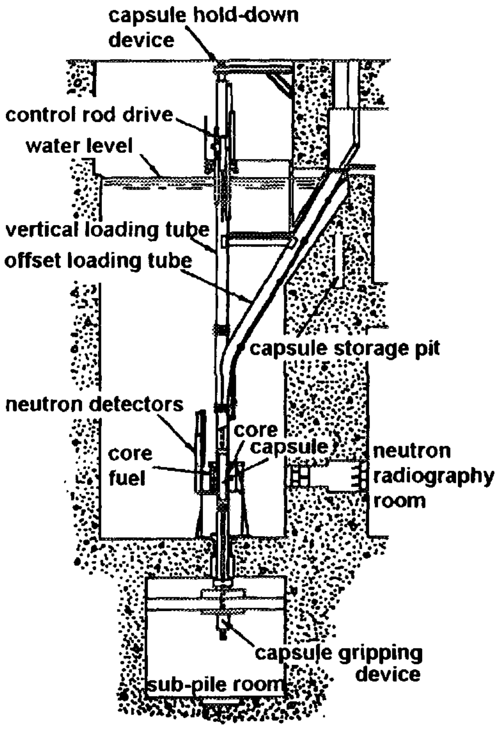


Figure 6 General arrangement of the NSRR

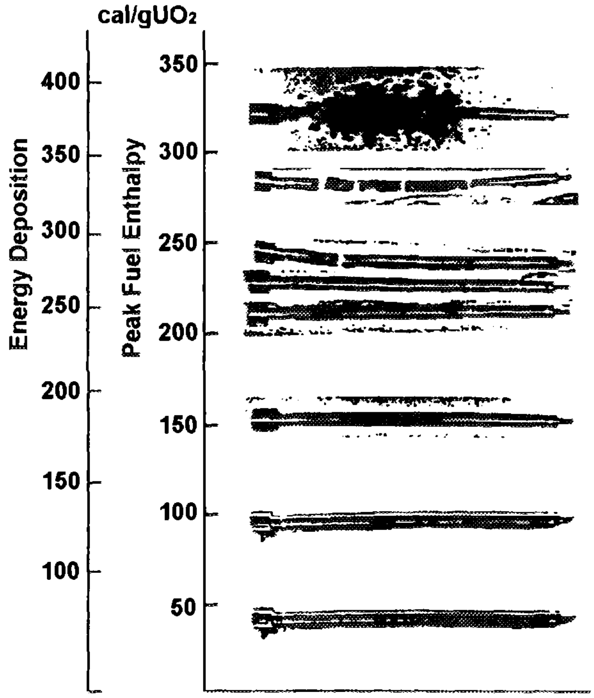


Figure 7 Appearance of post test fuel rods related with peak fuel enthalpy

7.2 Test Method

The NSRR is a modified TRIGA-ACPR (Annular Core Pulse Reactor). The general arrangements of the NSRR is shown in Fig. 6. The core structure is mounted on the bottom of 9 *m* deep open water pool. The pulsing operation is made by quick withdrawal of transient control by pressurized air. The maximum peak reactor power is about 21,000 *kW* in about 4 *msec*.

In typical tests (Standard tests), a test fuel rod is installed in a stainless steel capsule in the stagnant water at ambient pressure. The capsule and fuel rod are heavily instrumented, for example, by cladding thermocouples, fuel centerline thermocouple, strain gauges on the cladding, pellet/cladding elongation gauges, fuel internal pressure sensor and so on.

Tests in the pressurized conditions and under fuel bundle conditions have been conducted. Observations capsule with high speed camera to record the fuel outer appearance during tests have been developed and utilized.

7.3 Major Experimental Results with Fresh Fuel Rods

7.3.1 General Fuel Behavior

In order to study general failure mode of the fresh fuel (PWR type) standard tests rods were subjected to energy deposition of about 50 to 550 *cal/g* UO_2 or corresponding peak fuel enthalpies of about 30 to 450 *cal/g* UO_2. The photographs of typical post-tests fuel rods are shown in Fig. 7.

When the peak fuel enthalpy is below 88 *cal/g* of UO_2 DNB (Departure from Nucleate Boiling) does not take place and there is no change in appearance. After peak fuel enthalpy is beyond 110 *cal/g* UO_2 DNB is generated and cladding surface of the heated zone is oxidized. Fuel failure takes place at about 220 *cal/g* UO_2. Failure is caused by the embrittlement of the cladding due to oxidation and partial melting. When the peak fuel enthalpy exceeds 325 *cal/g* UO_2 fuel pellet melts and is ejected into water accompanying cladding failure. Molten fuel fragmentation into water causes generation of mechanical energy.

7.3.2 Influence of Fuel Design Parameters

The influence of major fuel design parameters, such as fuel internal pressure, gap sizes, gap gas composition, etc., on the fuel failure has been investigated. Among the parameters, the influence of fuel rod internal pressure was most significant. The data were already shown on Fig. 3. When the internal pressure is high, the failure mode changes form cladding melt to high temperature burst of the cladding.

7.3.3 Influence of Cooling Environment

Influence of bundle geometries, system pressure, cooling conditions, etc., on the fuel failure has been investigated. Under the bundle geometry failure threshold was found to be reduced by about 15% of the standard single rod test due to the reduction of heat conductance at the cladding surface. However, tests under simulated BWR and PWR cooling conditions gave similar failure threshold because the standard tests, as the cooling of the rod, were improved by forced cooling conditions.

7.3.4 Water Logged Fuel Failure

Water logged fuel generally fail when the fuel enthalpy reached about 100 *cal/g* UO_2 due to the burst of the cladding. In case of such failure, most of fuel pellets were released into water in fine particles and caused the generation of mechanical energy.

7.3.5 Generation of Mechanical Energy

When the fuel pellet is molten and ejected into water, molten fuel becomes fine particles and causes generation of mechanical energy by molten fuel/coolant interaction. The threshold enthalpy of this phenomena is found between 285 and 325 *cal/g* of UO_2.

7.4 RIA Behaviour of Irradiated PWR Fuel Rods

Experimental program with perirradiated LWR fuel rods as the test sample was started in 1989 in the NSRR. In this program transient behavior and failure initiation have been studied for PWR rods. Two types of rods were prepared. One is a commercial PWR fuel of 14×14 type (2.6% enriched, 95% TD) from power reactor and refabricated into segment rods (PWR-rods) at JAERI and the other is 10% enriched 14×14 type (95% TD) fuel segments (JM-rods) irradiated in JMTR of JAERI. The latter rods were used to study the fuel behavior at higher energy deposition. The peak fuel enthalpies for the former and the latter rods ranged 55 - 104, 96 - 164 *cal/g* UO_2, respectively.

In the PWR-rods, creepdown have taken place during irradiation and during the pulse irradiation in NSRR large PCMI was observed in comparison with the fresh fuel rods. Concerning JM-rods large deformation took place for the rods with fuel enthalpy above 200 *cal/g* UO_2. Fig. 8 shows the Ceramographs of the failed rod which is composed of 5% and 10% enriched pellets. Large diameter change occurred on 10% enriched pellet, due to swelling and crack in the pellet.

Fig. 9 shows fission gas release (FGR) from PWR-rods. From figure it is clear that FGR rate became larger when energy deposition was 113 (peak fuel enthalpy 85 *cal/g* UO_2). The cross section of the specimen revealed large number of hair cracks in the periphery of the pellet and it was considered that these hair cracks played important role in the fission gas release.

To summarize the results, energy depositions of each experiment are plotted as a function of fuel burnup and compared with all the available data from SPERT and PBF experiments in Fig. 10. Solid dots indicate fuel failure. More data will be produced in high burnup region in the future.

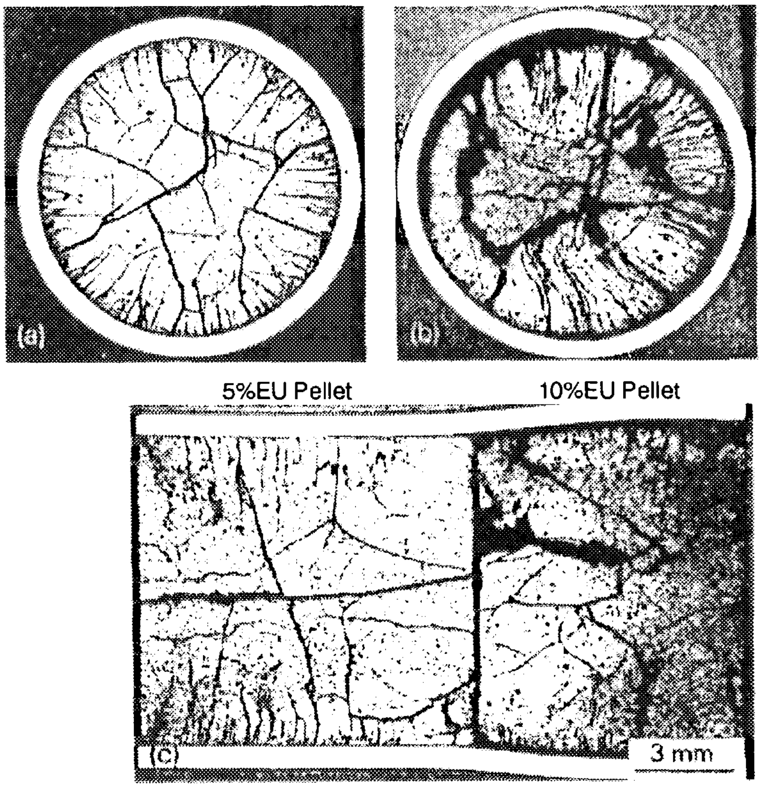


Figure 8 JM-rods after pulse irradiation. Cross sections at energy deposition of 113 cal/g (a) and 219 cal/g (b) and longtitudinal section at 219 cal/g (c).

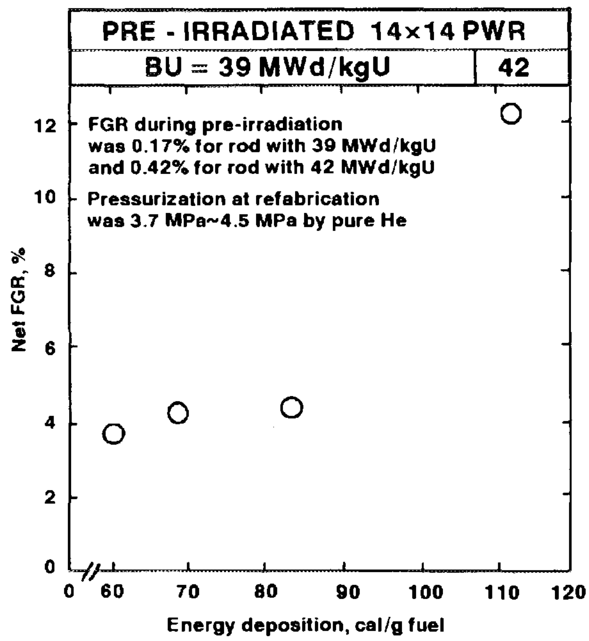


Figure 9 Relation of fission gas release and energy deposition of 14×14 PWR rod [3].

8 Application of the NSRR Results to the Licensing Giude

On the basis of the NSRR experiments, the Nuclear Safety Commission of Japan established a guide as stated above in 1984. For an anticipated transients the guide requires the acceptance design limit which is shown in Fig. 3 in dotted line to estimate number of failed rods. The acceptance design limit has been determined based on the single fuel rod test data with unpressurised and pressurized PWR type rods as shown in Fig. 3 in solid line. The difference between them include 16% reduction of failure threshold due to rod bundle effect and 10 *cal/g* UO_2 as a margin. The lowest bound of 65 *cal/g* UO_2 comes from the threshold enthalpy for the onset of DNB, as the high temperature burst of the cladding does not take place unless DNB occurs.

Concerning the generation of mechanical energy the highest peak fuel enthalpy at which fragmentation did not occur in the NSRR single rod test was 285 *cal/g* UO_2. The maximum fuel enthalpy for mechanical energy release as determined in the guide (230 *cal/g* UO_2) has been decided by reducing 15% and 10 *cal/g* UO_2 following the same manner as taken in determining acceptance fuel design limit in the case of anticipated transients.

Concerning the failure of water logged fuel, the highest fuel enthalpy at which rapture of the water logged fuel did not occur in the NSRR single rod tests was 90 *cal/g* UO_2. From this result the rapture threshold of the water logged fuel of 65 *cal/g* of UO_2 in the guide has been derived with reduction of 15% an 10 *cal/g* UO_2.

9 Concluding Remarks

In the present paper the general aspects of regulatory guides and related committee reports which are closely related PWR furl are described. Concerning the reactor safety research only the experiments at NSRR are described in connection with the guide for reactivity initiated events.

In Japan nuclear safety research, including reactor safety research has been conducted to support the regulatory activities by establishing/revising guides and preparing the necessary regulatory data, as well as improving nuclear safety. Nuclear safety research in Japan has been conducted in accordance with the national five years annual program.

Concerning LWR fuel safety research, fuel behavior in RIA and fuel integrity in normal operating conditions are major subjects of studies [4]. Perhaps it should be emphasized that fuel integrity has been the subject of safety research in Japan.

More efforts are needed to conduct nuclear safety research to ensure and to promote nuclear safety including reactor safety.

References

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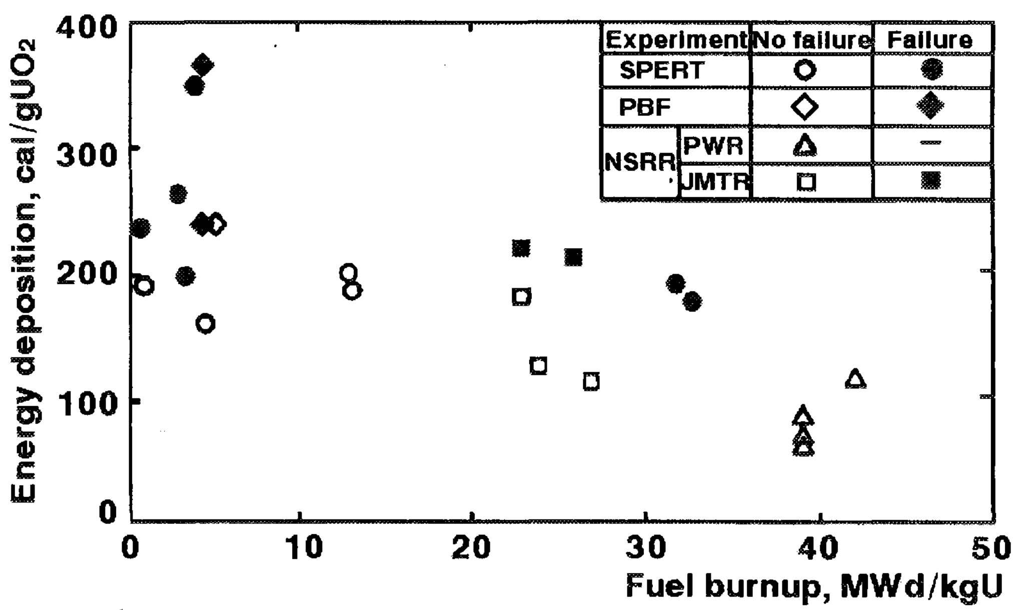

Figure 10 Results of SPERT, PBF and NSRR experiments with pre-irradiated fuels [3].