Cladding Performance under Power Oscillations in BWRs

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Abstract

Stability has been the main concern in the development stage of BWRs in history. Although stability studies of BWRs have a long history and there is extensive literature available, the literature on fuel performance under unstable power oscillations is very limited. It is very desirable to investigate the fuel integrity in BWR power oscillation conditions.

This goal of this work is to examine the cladding performance under the power oscillations. The calculations of fuel deformation were first conducted by using the FRAPTRAN code to simulate the Japanese FK11 test and power oscillation under normal BWR operating conditions. Then, based on the results of FRAPTRAN, the cladding fatigue analysis was conducted.

Based on the analysis, it could be concluded that: ① FRAPTRAN code could be used to analyze fuel performance under BWR power oscillations. The FRAPTRAN code results were in good agreement with FK-11 test in Japan. ②The fuel deformation was mainly caused by PCMI and was roughly proportional to the fuel enthalpy. Enhanced cladding deformation due to ratcheting was not found. ③The cladding could satisfy the design criterion in ASME Code under power oscillation conditions, which means it could maintain the fuel integrity. ④Cladding thermal fatigue is not an issue under power oscillations, unless dryout takes place.

1. Introduction

1.1 BWR Stability and its Effects to Safety

In historical perspective, stability has been the main concern in the development stage of BWRs. It is of practical importance for designing and operating BWRs.

The boiling two-phase flow in the core may become less stable because of the time lag between vapor generation and pressure loss perturbation. Furthermore, in BWRs, the reactivity depends strongly on the core void fraction. Thus, when a void fraction oscillation is established in a BWR, the power oscillates according to the neutronic feedback. This feedback mechanism may under certain conditions lead to poorly damped or even limit-cycle power oscillations. Their frequency lies around 0.5 Hz (about twice the transport time of the coolant through the core). Amplitudes from nearly 0% to more than 100% in power have been observed. The oscillations are mostly global, i.e. "in-phase". [Hanggi P., 2001]

Higher mode power oscillations are also possible; these divide the core into two regions oscillating in opposite directions at constant overall power. These regional oscillations, also called "out-of-phase" oscillation, are cumbersome for the operators since their detection is not directly possible with standard instruments that display only core-average data.

Generally, the BWR stability issue is not a major industry safety problem from a technical point of view. Given appropriate instrumentation, power oscillations are easy to detect and there exist simple, as well as effective, counter measures. A scram will normally solve the problem. But the concerns of out-of-phase oscillation mode and the unavailability of reactor shutdown system (the case of ATWS originated oscillations) keep the stability issue a safety concern.

In terms of safety, the concerned variables in an instability occurrence with high oscillation amplitudes are the neutron flux and the rod surface temperature: the control of the first of the above quantities may prevent any undesired excursion of the second one. Additional problems might arise due to thermal cycling that may affect the fuel rod integrity, making pellet-cladding interaction more probable; thermal cycling may also induce greater than normal fission product release from pellets. [F. D'Auria, 1997]

The numerous modifications in reactor size, reactor power, fuel design, power density, discharge burnup and loading strategies changed the core stability behavior of the BWR reactor to a significant extent. Since, for economic reasons, the trend towards smaller-diameter fuel rods and different loading strategies will persist, the stability problem has to be taken care of. It is, therefore, important to understand the underlying mechanism of power-void instability as thoroughly as possible, as well as to be able to understand its consequence to fuel integrity.

In literature, there are numerous works on BWR stability. But most of them are focused on knowing the mechanism of different of oscillations, and on how to predict the oscillations by developing all kinds of codes. Very limited work was done on the effects of oscillation.

This work is focused on if the fuel integrity (failure of cladding) could be conserved under the power oscillations.

1.2 Fuel Failure Modes

An LWR fuel rod typically consists of UO2 fuel pellets enclosed in Zircaloy cladding. The primary function of the cladding is to contain the fuel column and the radioactive fission products. If the cladding does not crack, rupture, or melt during a reactor transient, the radioactive fission products are contained within the fuel rod. During some reactor transients and hypothetical accidents, however, the cladding may be weakened by a temperature increase, embrittled by oxidation, or over stressed by mechanical interaction with the fuel. These events alone or in combination can cause cracking or rupture of the cladding and release of the radioactive products to the coolant. Furthermore, the rupture or melting of the cladding of one fuel rod can alter the flow of reactor coolant and reduce the cooling of neighboring fuel rods. This event can lead to the loss of a "coolable" reactor core geometry.

The primary means of cladding failure result from mechanical loading of the internal clad surface by fuel-clad interactions or from the buildup fission product gasses in the plenum to critical values. Failure in both cases is due to rupture from plastic deformation and creep. Analysis of the mechanical behavior is a very complicated task due to the complicated relationships which exist between the mechanical, thermal, chemical, and structural properties. Additionally, not all of the individual mechanisms involved are completely understood. [MIT 22.314 Lecture Notes F.1]

The most prevalent failure mode for fuel rods is due to the failure of cladding from Pellet-Cladding Mechanical Interactions (PCMI). PCMI failures are the result of mechanical interaction between the fuel and cladding in a corrosive environment. Thermal expansion mismatches between the fuel pellets and cladding material result in pellet expansion exceeding that of the clad. This internal stress is opposed by the external stress of the pressurized cooling system such that a tight interface between the two materials is maintained. Volatile fission products of cesium and iodine released to the gap contribute a corrosive environment to the interface and result in Stress Corrosion Cracking (SCC) of the interior surface of the cladding. Thermal shocks induced in the clad by power ramping establish stress gradients in the clad and further the contact stress caused by the mismatch in thermal expansion coefficients resulting in crack advancement through the cladding until the cladding is eventually breached. The geometric shape of the fuel also changes as burnup proceeds. Cracks in the UO2 matrix develop which can result in the formation of sharp points and edges on the exterior surface of the fuel. During pellet-clad interactions, these features act to amplify the local stress to the clad and can result in premature failure of the rod. [Frost, Brian R. T., 1982]

Fuel element failure may also occur due to the process of hydriding of the zircaloy cladding. Zircaloy has a high affinity for hydrogen and oxygen with a preference for interaction with oxygen to form zirconium oxide. On the external surface, a thin oxide film is formed by interaction of the clad with the coolant which provides a protective coating and an effective barrier to external hydriding. The internal surface of the clad is, however, subject to hydriding. Internal sources of hydrogen include: helium gas impurities, radiolytic decomposition of organic contaminants, hydrogen trapped in pores of the UO2 lattice and moisture absorbed in the fuel pellets following fabrication and prior to assembly of the fuel pins. Hydrogen is absorbed into the zircalloy ensuing in the formation of zircalloy hydride, which is less dense and more brittle than the original material. Stresses are established which nucleate blisters and cracks in the cladding which continue to grow under the formation of additional hydrided material until the cracks completely penetrate the clad. [Ygnik S. K., 1993]

Failure of fuel rods may also occur as the result of wear at contacts points between the cladding and the spacer grid caused by flow induced vibrations. Design modifications have effectively eliminated the occurrence of these events by ensuring the use of retainer springs with sufficient strength to minimize vibration. During operation, fuel elements tend to warp and bow from their original cylindrical configuration. These effects have not proven to be a problem in the past but are a concern for higher burn-up fuels which will experience these forces over longer time periods and can be expected to result in even more distortion. [Frost, Brian R. T., 1982]

However, in case of power oscillation, the major effect is the cyclic thermal load and its consequence. Thus, our major concern is its effects on pellet-cladding mechanical interaction, fission gas release, and the thermal fatigue of cladding in long time oscillations.

1.3 Literature Review of Fuel Performance under Power Oscillations in BWRs

Although instability studies of BWRs have a long history and there is extensive literature available, the literature on fuel performance under unstable power oscillations is very limited.

In order to examine high burnup fuel performance under power oscillation conditions, two sets of irradiated fuels under simulated power oscillation conditions were conducted in the Nuclear Safety Research Reactor in Japan (Nakamura etc., 2003). Impacts of cyclic loads on the fuel performance under hypothetical unstable power oscillations arising during an anticipated transient without scram (ATWS) in BWRs were examined in the tests. Deformation of the fuel cladding of the rest rods was comparable to those observed in shorter transient tests, which simulated reactivity-initiated accidents (RIAs), at the same fuel enthalpy level. It was concluded that the fuel deformation was mainly caused by PCMI and was roughly proportional to the fuel enthalpy. Enhanced cladding deformation due to ratcheting by the cyclic load was not observed. Fission gas release, on the other hand, was considerably smaller than in the RIA tests, suggest different release mechanisms in the two types of transients.

Takanori Fukahori, etc. (2005) developed an analysis system code, TRUST, for fuel integrity during hypothetical core instability events in BWR cores. This system can estimate the thermal-hydraulic and mechanical properties such as fuel cladding temperature, MCPR, rod internal pressure. Based on their systematic analysis, it is shown that fuel integrity could be maintained even if the neutron flux oscillation would be large enough to exceed the scram level. However, according to their analytical results, the pellet-cladding mechanical interaction was not predicted during core-wide oscillation. By the boiling transition during the core-wide oscillation, the cladding temperature was up to about 780K, but the cladding oxidation was negligible. At the peripheral region of pellet, the temperature exceeded the temperature under the rated operation by boiling transition, but fission gas release was not significant.

The limited available literature also encouraged this term paper on fuel integrity under the power oscillations.

1.4 Objectives of this Work

This goal of this work is to examine the fuel integrity under the power oscillations. The calculations of fuel deformation were first conducted by using the FRAPTRAN code.

Then, based on the results of FRAPTRAN, the cladding fatigue analysis was conducted. The work includes:

(1) Simulating the power oscillation tests of the NSRR (FK11) by using FRAPTRAN code, and make a comparison with experiment results.
(2) Simulating the power oscillations without scram in normal BWR operating conditions.
(3) Analyzing the influence on cladding fatigue during power oscillations.

2. Description of FRAPTRAN Code

2.1 Objectives and Scope of the FRAPTRAN Code

FRAPTRAN (Fuel Rod Analysis Program Transient) is a FORTRAN language computer code developed for the U.S. Nuclear Regulatory Commission to calculate the transient thermal and mechanical behavior of light-water reactor fuel rods. FRAPTRAN will be applied for the evaluation of fuel behavior during reactor power and coolant transients such as reactivity accidents, boiling-water reactor power oscillations without scram, and loss-of-coolant-accidents up to burnup levels of 65 GWd/MTU. The FRAPTRAN code is the successor to the FRAP-T (Fuel Rod Analysis Program-Transient) code series developed in the 1970s and 1980s. FRAPTRAN is also a companion code to the FRAPCON-3 code developed to calculate the steady-state high burnup response of a single fuel rod. [NUREG/CR-6739]

FRAPTRAN is an analytical tool that calculates LWR fuel rod behavior when power and/or coolant boundary conditions are rapidly changing. This is in contrast to the FRAPCON-3 code that calculates the time (burnup) dependent behavior when power and coolant boundary condition changes are sufficiently slow for the term “steady-state” to apply. FRAPTRAN calculates the variation with time, power, and coolant conditions of fuel rod variables such as fuel and cladding temperatures, cladding elastic and plastic stress and strain, and fuel rod gas pressure. Variables that are slowly varying with time (burnup) such as fuel rod densification and swelling, cladding creep and irradiation growth, and fission gas release, are not calculated by FRAPTRAN. However, the state of the fuel rod at the time of a transient, which is dependent on those variables not calculated by FRAPTRAN, may be read from a file generated by FRAPCON-3 or manually entered by the user. [NUREG/CR-6739]

The FRAPTRAN code has the capability of modeling the phenomena which influence the performance of fuel rods in general and the temperature, embrittlement, and stress of the cladding in particular. The code has a heat conduction model to calculate the transfer of heat from the fuel to the cladding, and a cooling model to calculate the transfer of heat from the cladding to the coolant. The code has an oxidation model to calculate the extent of cladding embrittlement and the amount of heat generated by cladding oxidation. A mechanical response model is included to calculate the stress applied to the cladding by the mechanical interaction of the fuel and cladding, by the pressure of the gases inside the rod, and by the pressure of the external coolant.

2.2 Cladding Deformation Model

The detailed description of cladding deformation model used in FRAPTRAN code could be found in Report NUREG/CR-6739. Here the author only discussed the different models in the strain-stress calculation in case whether the cladding-fuel gap was open or closed, which resulted in different mechanisms of deformation.

Deformation and stresses in the cladding in the open gap regime are calculated using a model which considers the cladding to be a thin cylindrical shell with specified internal and external pressures and a prescribed uniform temperature.

Calculations for the closed gap regime are made using a model which assumes that the cladding is a thin cylindrical shell with prescribed external pressure and a prescribed radial displacement of its inside surface. The prescribed displacement is obtained from the fuel thermal expansion model. Furthermore, because no slippage is assumed to take place when the fuel and cladding are in contact, the axial expansion of the fuel is transmitted directly to the cladding. Hence, the change in axial strain in the shell is also prescribed.

The decision as to whether or not the fuel is in contact with the cladding is made by comparing the radial displacement of the fuel with the radial displacement that would occur in the cladding due to the prescribed external (coolant) pressure and the prescribed internal (fission and fill gas) pressure. The decision is expressed by the equation:

$$u_r^{fuel} \geq u_r^{clad} + \ddot{a} \qquad (2.1)$$

where: $\ddot{a}$ = as-fabricated fuel-cladding gap size (m)

u_r = radial displacement (m)

If the above equation is satisfied, the fuel is determined to be in contact with the cladding. The loading history enters into this decision by virtue of the permanent plastic cladding strains imposed in the cladding by the cladding loads.

If the fuel and cladding displacements are such that Equation 2.1 is not satisfied, the fuel-cladding gap has closed during the current loading step and the open gap solution is used. If Equation 2.1 is satisfied, however, the fuel and cladding have come into contact during the current loading increment. At the contact interface, radial continuity requires that:

$$u_r^{clad} = u_r^{fuel} - \ddot{a} \quad . \tag{2.2}$$

while in the axial direction the assumption is made that no slippage occurs between the fuel and cladding. This state is referred to as PCMI.

Note that only the additional strain which occurs in the fuel after PCMI has occurred is transferred to the cladding. Thus, if $\varepsilon_{z,o}^{clad}$ is the axial strain in the cladding just prior to contact and $\varepsilon_{z,o}^{fuel}$ is the corresponding axial strain in the fuel, then the no-slippage condition in the axial direction becomes:

$$\varepsilon_z^{clad} - \varepsilon_{z,o}^{clad} = \varepsilon_z^{fuel} - \varepsilon_{z,o}^{fuel} \tag{2.3}$$

After u_r^{clad} and ε_r^{clad} have been calculated, a solution is made of the stresses and strains in a thin cylindrical shell with prescribed axial strain, external pressure, and prescribed radial displacement of the inside surface. The solution also gives the interface pressure between the fuel and cladding.

2.3 Fuel Rod Internal Gas Pressure Response Model

The pressure of the gas in the fuel rod must be known in order to calculate the deformation of the cladding and the transfer of heat across the fuel-cladding gap. The pressure is a function of the temperature, volume, and quantity of gas. The detailed description of cladding deformation model used in FRAPTRAN code could be found in Report NUREG/CR-6739.

It should be mentioned here that FRAPTRAN does not have a model to calculate the transient release of fission gases as a function of temperature. The fill gas composition and pressure at the time of the transient, which is dependent on fission gas release prior to the transient, is either manually entered by the user or read from a FRAPCON-3 burnup initialization file.

The released fission gas does affect the gas pressure and composition, which in turn impacts the transient thermal and mechanical calculations. It has also been proven that some amount of fission gas was released during power oscillation transients. Therefore, a user input option (MODEL data block in the input file) is used to specify the fission gas release to the fuel-cladding gap and rod plenum during power oscillation transients.

Although the rod-average fractional fission gas release could be specified as a function of time during the transient, we don't know the detailed information about the fission gas release in a power oscillation test. For simplicity, the internal gas pressure, rather than fission gas release fraction, was specified as a function of time. In our analysis, different internal gas pressure history was used to identify what is the effect of this simplification.

3. Cladding Failure Criterion

The criterion is decided based on two documents: NRC NUREG-0800, Standard Review Plan, and ASME B&PV Code Section III Article NH-3000.

3.1 Cladding Stress Criterion

Where creep is significant, the ASME B&PV Code Section III Article NH-3000 specifies that the strain limiting criteria, rather than stress limiting criteria are applied. However, simplified methods can be used to establish conservative limits for stress.

(1) Sm = min. of {2/3 Sy at ambient (room) temperature, 2/3 Sy at service temperature, 1/3 Su at ambient (room) temperature, 1/3 Su at service temperature}

(2) St = min. of {100 % of the stress to cause 1% strain, 80% of the stress to initiate tertiary creep, 67% of the minimum stress to cause rupture}

The time independent stress limits for the load categories are as follows with: Pm, primary membrane stress (pressure difference across the cladding and PCMI); Pl primary local stress (stress raiser due to pellet cracking and bambooing); Pb, primary bending stress (bowing or PCI gradients); and Q, secondary stress (thermal stresses)

- **Pm < 1.0 Sm**
- **Pm + Pl < 1.5 Sm**
- **Pm + Pl + Pb < 1.5 Sm**
- **Pm + Pl + Pb+ Q < 3.0 Sm**

The stress adder Q is included to assure that the transient thermal stresses do not exceed stresses which could exhaust the deformation capability of materials

It should be noticed that typical fuel performance codes (like FRAPCON) calculate Pm and Q, but not Pl and Pb.

3.2 ***Cladding Strain Criterion***

The total permanent uniform strain shall not exceed:

- 1 % membrane strain (limiting)
- 2 % bending strain
- 5 % local strain

The intent of this requirement is to limit cladding damage due to slow rate strain accumulation at which the stress does not reach the stress limit (yield stress). The clad loading mechanism is the rod internal differential pressure with the system pressure and clad straining by the pellet expansion and PCMI. A bending strain and local strain are not calculated by FRAPCON and the limits are not applied at this time.

4. Simulating the Power Oscillation Test FK11

4.1 FK11 test procedure and test conditions

In order to examine high burnup fuel performance and under unstable power oscillation conditions arising during an anticipated transient without scram (ATWS) in boiling water reactors, fuel irradiation tests were conducted with irradiated fuels under the simulated power transient conditions in the Nuclear Safety Research Reactor (NSRR), Japan.

The power oscillation tests for BWR fuel rods in NSRR is also called test FK-11. In FK-11 test, irradiated fuels at burnups of 56 GWd/tU were subjected to four power oscillations, which peaked at 50 at intervals of 2 s. Peak fuel enthalpies were estimated to be 256 J/g (61 cal/g) in the test. The power oscillation was simulated by quick withdrawal and insertion of six regulating rods of the NSRR. The detailed fuel design and pre-test conditions are showed in the table below.

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Power history in test FK-11 is illustrated in Fig. 4.1. At the beginning of the test, rod power was kept at 27 kW/m for about 5 s to develop a center peaked temperature profile before initiating the power oscillation transient. Then, the test rod was subjected to four power oscillations to the peaks at 38, 43, 47, 50 kW/m at intervals of 2 s. Fuel enthalpy during the test relative to room temperature was estimated with FRAP-T6 and is illustrated in Fig. 4.1.

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However, the FRAPCON steady-state output file, which is used as an input for FRAPTRAN, is prepared based on Fukushima Diana 2 Reactor, also known as base irradiation file for FK1. There are some differences between FK1 and FK11 pre-test conditions, which are shown in the table below.

Table 4.2 Difference between FK-11 pre-test and FRAPTRAN input condition

	Parameter	Unit	FK-11 pre-test	FRAPTRAN input
Design	Fill Gas Pressure	MPa	0.5	0.3
	Fuel Density	%	97	95
	Fuel Enrichment	%	4.5	3.9
Irradiation	Peak Linear Power	kW/m	35	17.3
	Irradiation Time	day	4 cycles	2500
	Burnup	MWd/kg	56	65
	EOL Gas Pressure	MPa	1.4	0.6

The major differences between the pre-test condition and the FRAPCON simulation results may be the burnup and the internal gas pressure. As in our FRAPTRAN input file, the internal gas pressure was specified as a user input, we could changed the initial gas pressure to 1.4 MPa, the same as FK-11 pre-test condition. The difference of burnup may affect the results in some degree, so our comparison between the test and simulation could only be qualitative. It should be also mentioned that the FK-11 test is operating under cold zero conditions before power oscillation starts. Thus, we need to further conduct analysis in normal BWR operating conditions to investigate if the fuel integrity could be maintained under power oscillations.

4.2 Results of tests and FRAPTRAN simulation

In test FK-11, synchronized axial elongation of fuel stack and cladding was observed as shown in Fig. 4.2. This deformation behavior suggests that strong PCMI controlled the cladding deformation during the test. Post-test diameter measurement showed that both radial and axial deformation was elastic. The cladding surface temperature remained about 100ºC throughout the power transient. Rod internal pressure increased gradually from 1.4 to 2.2 MPa during the power transient, a result of heat up of the gap/plenum gas and fission gas release from the pellets. Post-test gas analysis indicated that fission gas release during the test was 0.4% of the total produced in the pellets. Test results are summarized in Table 4.3

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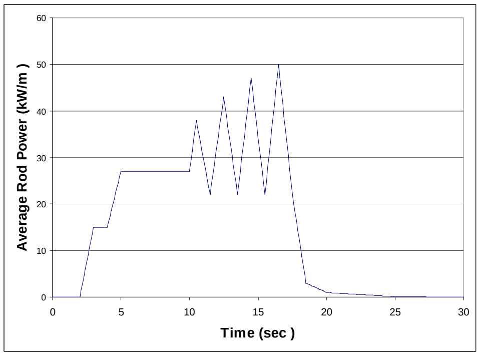
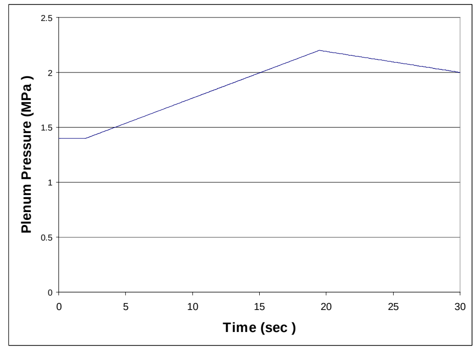
Table removed due to copyright restrictions.

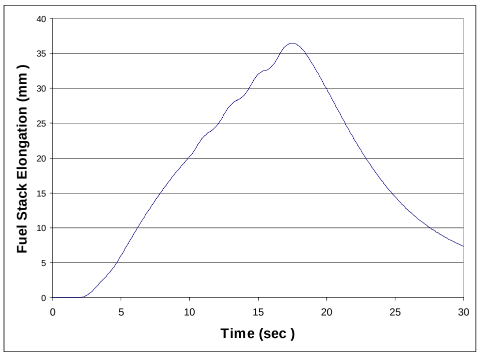
After changing the FRAPTRAN input file to satisfy the FK-11 test conditions, FRAPTRAN 1.3 Code was ran, and the results was shown in the Fig. 4.3 and Fig. 4.4.



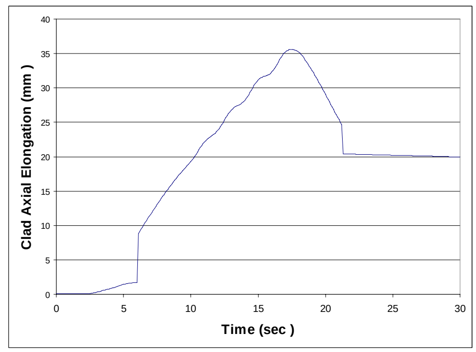
(a) Average Rod Power History



(b) Internal Gas Pressure Histroy

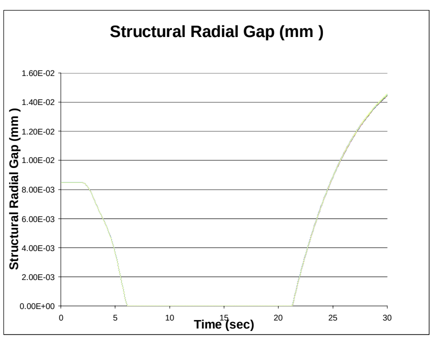


(c) Fuel Stack Axial Elongation

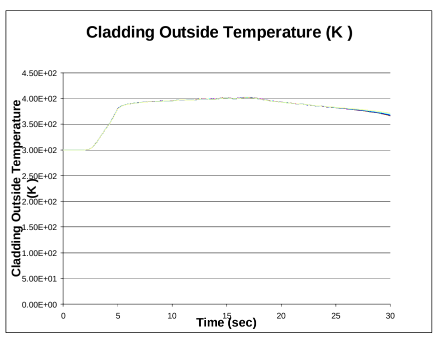


(d) Clad Axial Elongation

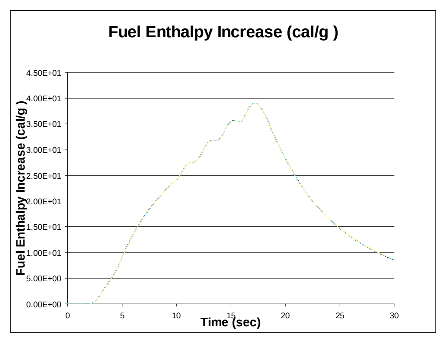
Fig. 4.3 Results of elongation of pellet stack and cladding, rod internal pressure and reactor power of FRAPTRAN simulation for test FK-11



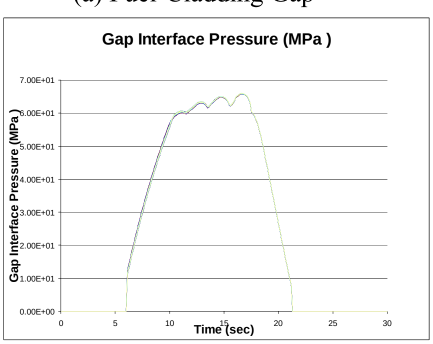
(a) Fuel-Cladding Gap



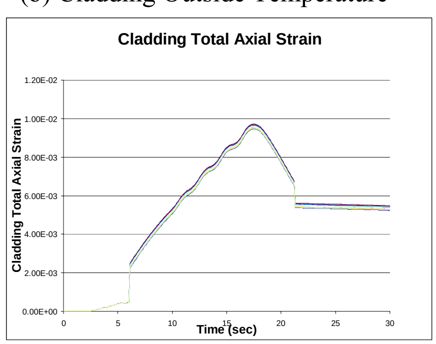
(b) Cladding Outside Temperature



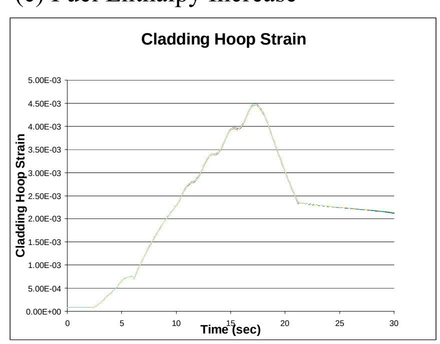
(c) Fuel Enthalpy Increase



(d)Gap Interface Pressure



(e) Cladding Axial Strain



(f) Cladding Hoop Strain

Fig. 4.4 Results of other parameters of FRAPTRAN simulation for test FK-11

4.3 Discussion

From FRAPTRAN simulation results, we can find that the axial elongation of fuel stack and cladding was nearly the same in Fig. 4.3.c and 4.3.d; and also that the structural radial gap is zero in power oscillation periods (Fig. 4.4.a) and enormous amount of gap interface pressure (clad-pellet interaction, Fig. 4.4.d). These both suggest that strong PCMI controlled the cladding deformation in the oscillations, which is consistent with the test results.

However, the test results shows that very small plastic cladding deformation was left after the power oscillation, while the FRAPTRAN simulation results indicated fairly large plastic cladding deformation after the oscillations. Also the amplitude of the peak clad axial elongations between test and simulation are a little different. These differences may arise from the difference in initial burnups (56 MWd/kg in test, 65 MWd/kg in simulation).

Furthermore, from Fig.4.4.e and Fig.4.4.f, the cladding deformation was not accumulated through the power cycling, indicating that ratcheting deformation of the cladding by the pellets did not occur, which agree with the test results. It could also be shown that the fuel axial elongation is proportional to the fuel enthalpy increase from Fig. 4.3.c and Fig. 4.4.c., which also is consistent with the test results.

In FK-11 test, the cladding axial elongation is around 47mm, the corresponding axial strain is exceeding 1% membrane strain limits. However, the post-test examination showed that the cladding is not failed. In FRAPTRAN simulation, the cladding axial strain is around 0.8%, below the 1% membrane strain limit.

Although the existence of the small difference in initial conditions between FK-11 test and FRAPTRAN simulation, the results of them are in good agreement. This gives us some confidence to use FRAPTRAN code to simulate power oscillations in normal BWR condition, rather than cold zero condition in FK-11 test. It could be concluded that the fuel deformation in power oscillations is mainly caused by PCMI, and is roughly proportional to the fuel enthalpy, and enhanced cladding deformation due to ratcheting by the cyclic load was not observed (Also due to the assumption of non axial slippage at fuel-cladding interface in the model).

5. Power Oscillations under Normal BWR Operating Conditions

5.1 BWR Model

As the FK-11 test is operating under cold zero conditions before power oscillations, the cladding outside pressure and temperature are considerably lower than normal BWR operating conditions, which makes its deformation of cladding different from those under normal BWR oscillations. Therefore, we need to further analyze the cladding deformation in normal undamped BWR power oscillations.

During the out-of-phase instability, half of the core rises in power while the power in the other half decrease to maintain an approximately constant total core power. In the tests described in [E. Gialdi et al., 1985], local power oscillations amplitude was as large as 70% while the average reactor power oscillated by only 12%. Since the automatic safety systems in BWRs rely on total power measurements to scram the reactor, large amplitude out-of-phase oscillations can occur without reactor scram. Also, the system adjusts flow from one half of the core to the other half while keeping the total flow rate almost constant. Therefore, it is necessary to design the reactors to avoid the out-of-phase instability problem. However, the recent trend of larger reactor core for economic concerns makes reactors more favorable to out-of-phase oscillation mode.

Based on the above discussion, the linear heat generation rates of fuel rods, which are undertaken the power oscillations, are modeled to oscillate between 30% and 170% of normal linear heat generation rate. The oscillating coolant mass flow rates were decided by satisfying the constant pressure drop across the reactor core (constant pressure drop boundary condition). Upon the coolant pressure, it was considered constant across the core as the pressure drop are considerably small comparing to the operating pressure, 7.2 MPa.

We are continuing to use the FRAPCON output file preparing for Fukushima Diana 2 reactor, which is a BWR/4 type, to conduct our FRAPTRAN transient analysis. The linear heat generation rate and flow rate are calculated based on typical BWR/4 data, needed information was collected in Table 5.1; the input power history and coolant flow rate are shown in Fig. 5.1.

Table 5.1 Needed BWR/4 information for Power and flow rate calculation

Parameter	Typical BWR/4
Core	
Thermal power	3293 MWt
Core flow rate	12915 kg/s
Power density	51 kW/liter
Equivalent core diameter	4.75 m
Vessel inner diameter	6.375 m
Operating pressure	7.2 MPa
Core pressure drop	0.15 MPa
Reactor inlet temperature	275 °C
Radial power peaking factor	1.4
Fuel assembly	
Assembly number	764
Fuel pin lattice	Square 8x8
Number of fuel pins per assembly	63

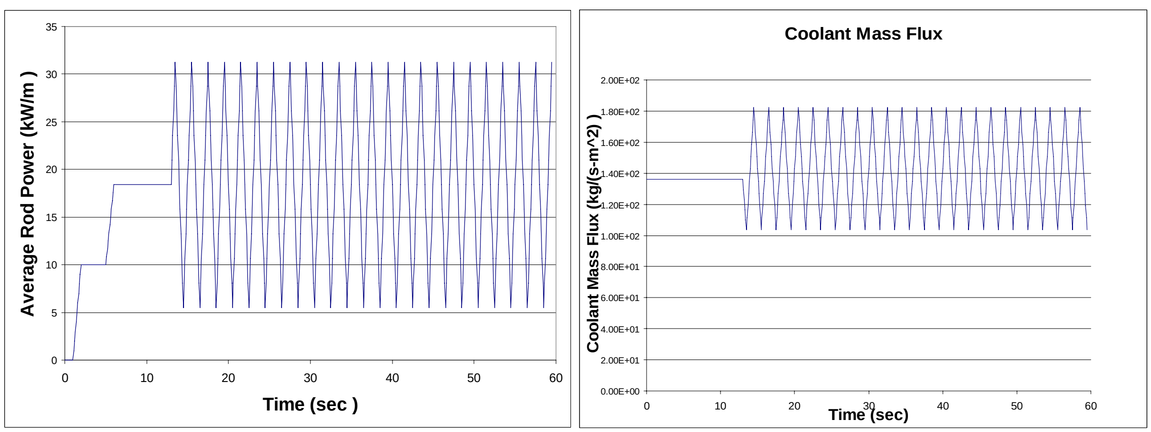
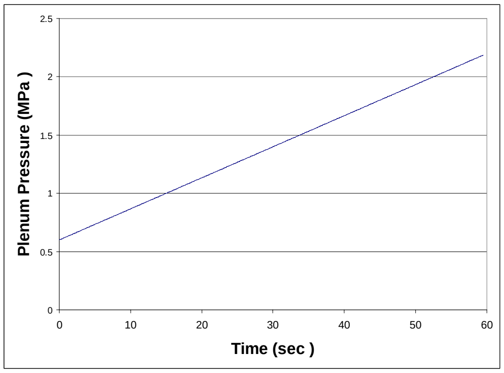


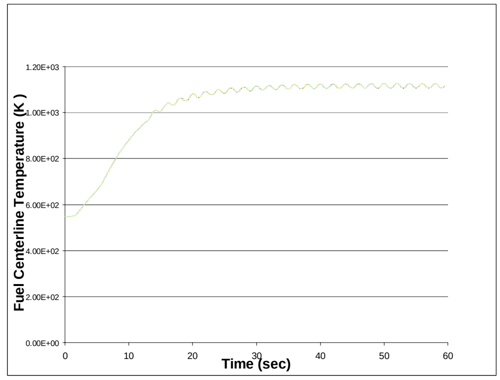
Fig. 5.1 History of linear heat generation rate and flow rate in power oscillations

5.2 Results of FRAPTRAN Simulation

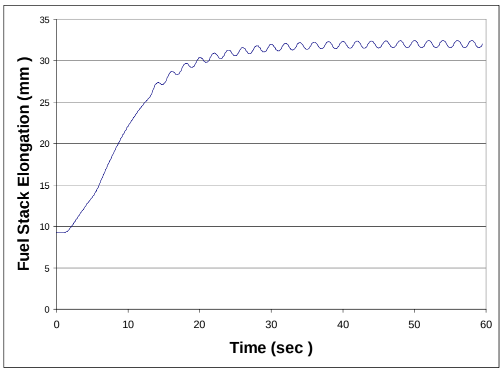
The major results of deformation of fuel and cladding under oscillations is shown in the Fig. 5.2, Fig. 5.3, and Fig. 5.4. It should be mentioned here that the fission gas release is not considered in FRAPTRAN code. Thus, as discussed in Section 2.3, the internal gas pressure was specified as a function of time in our analysis. The internal gas pressure history could be found in Fig. 5.2.a.



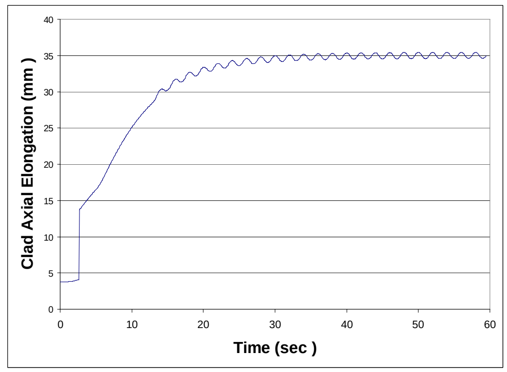
(a) Internal Gas Pressure Histroy



(b) Fuel Centerline Temperature

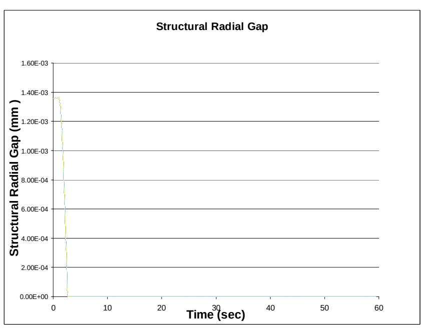


(c) Fuel Stack Axial Elongation

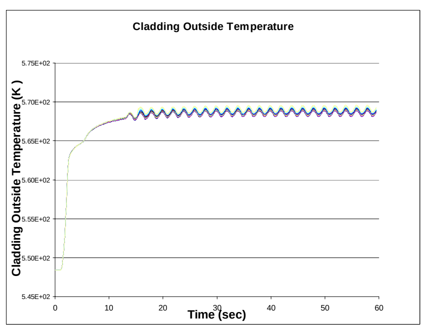


(d) Clad Axial Elongation

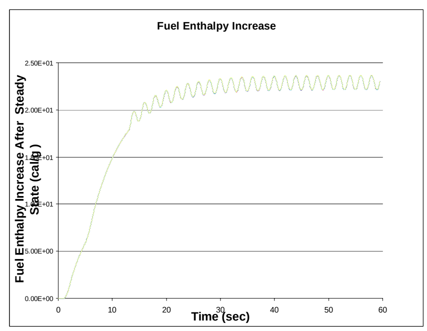
Fig. 5.2 Results of FRAPTRAN simulation for BWR power oscillation (1)



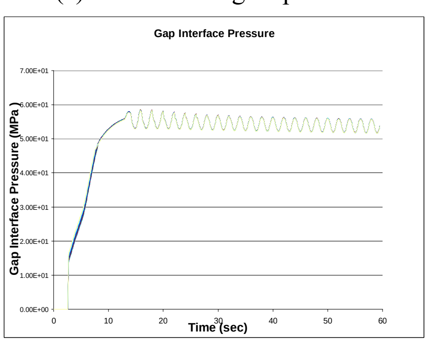
(a) Fuel-Cladding Gap width



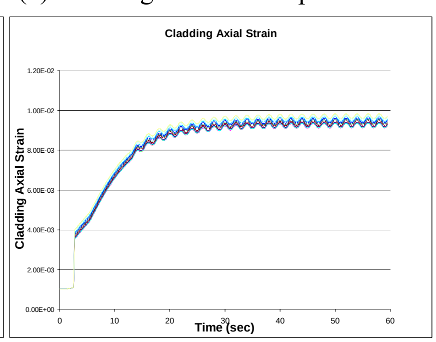
(b) Cladding Outside Temperature



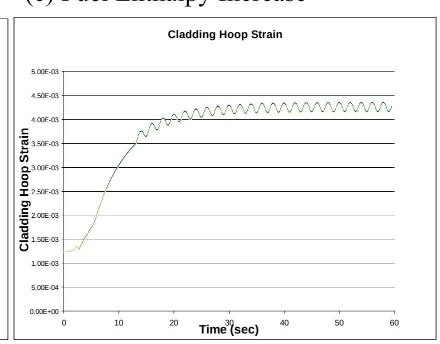
(c) Fuel Enthalpy Increase



(d)Gap Interface Pressure

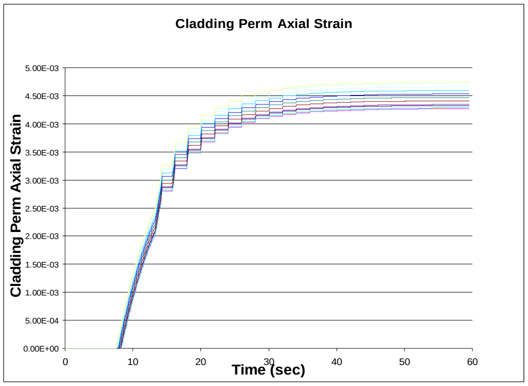


(e) Cladding Axial Strain

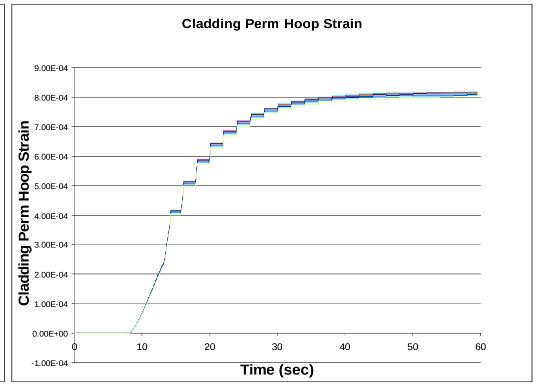


(f) Cladding Hoop Strain

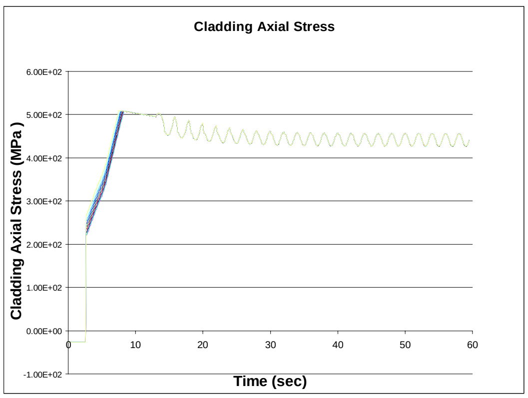
Fig. 5.3 Results of FRAPTRAN simulation for BWR power oscillation (2)



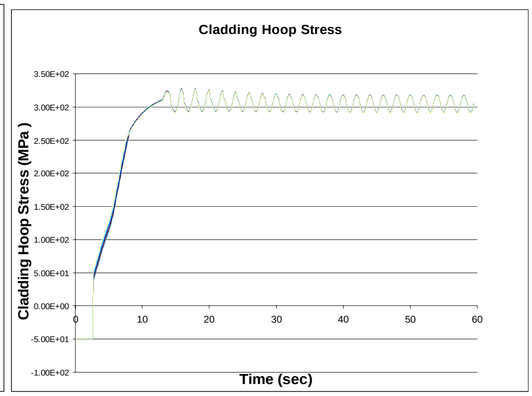
(a) Cladding Permanent Axial Strain



(b) Cladding Permanent Hoop Strain



(c) Cladding Axial Stress



(d) Cladding Hoop Stress

Fig. 5.4 Results of FRAPTRAN simulation for BWR power oscillation (3)

In Fig. 5.2.c and Fig. 5.2.d, cladding axial elongation follows quite well with the fuel pellet axial elongation; also Fig. 5.3.a and Fig. 5.3.d indicate the gap is closed under power oscillations. These facts suggest the strong PCMI controlled cladding deformation during oscillations.

5.3 Discussion

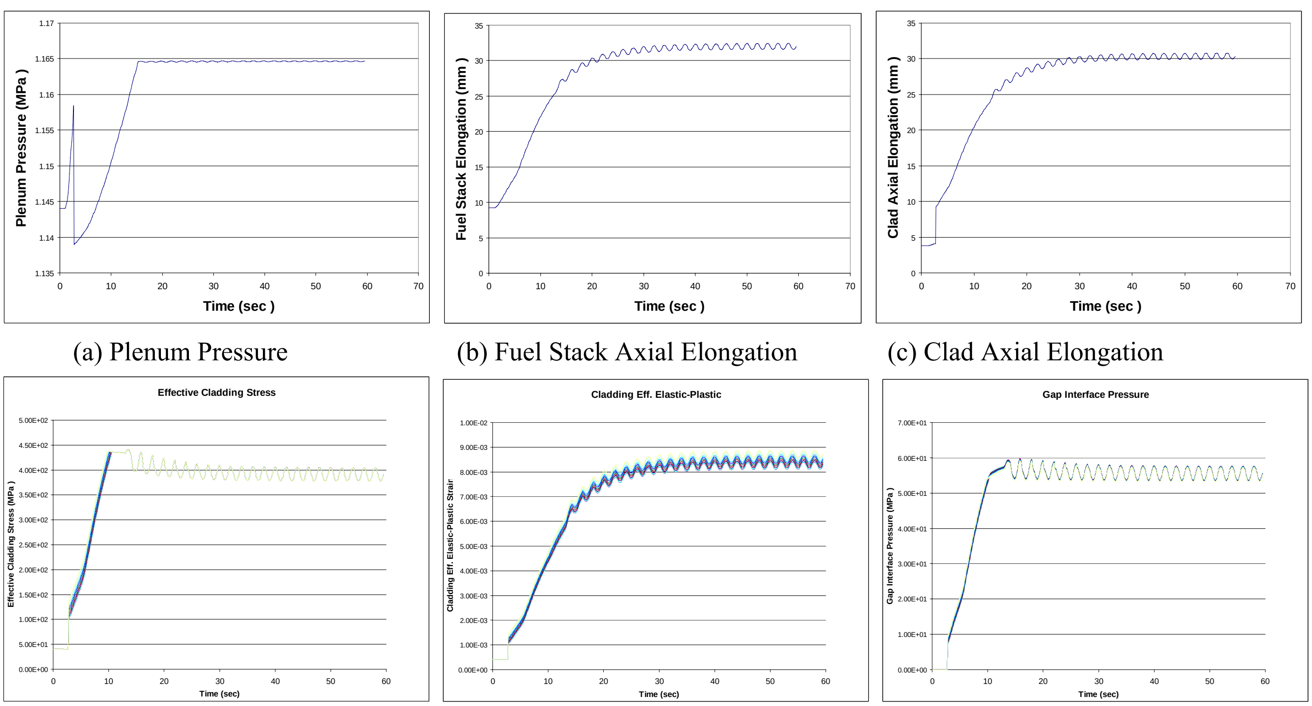
In this section, we followed the limiting criteria discussed in Section 3 to examine if the cladding is failed under power oscillations. The results were shown in Table 5.2.

Table 5.2 Simulation results applying to the design criterion

Design Criterion	Simulation Results	Criteria Values	If Satisfy
Pm< 1.0 Sm	$Pm = \sigma_{tresca} = \lvert \sigma_\theta - \sigma_r \rvert$, $Max(Pm) = 300MPa$	$Sy \approx 500MPa$, $Sm = 2/3Sy = 333MPa$	Yes
Pm+Q<3.0 Sm	$Max(Q) = 160MPa$, $Pm + Q = 460MPa$	$3Sm = 1000MPa$	Yes
1% membrane strain	$Max(\varepsilon_{eff}) = 0.0103$ $Overall, \varepsilon_z = 0.008$	1%	No

It should be mentioned that the thermal stress Q achieved its maximum value when the fuel rod increased its power form zero to rated power before power oscillation started, which may not happen in the real plants. Although the 1% membrane strain criteria is not satisfied, we still have enough confidence that the fuel rod would not fail as we can see the cladding strain and permanent axial and hoop strain did not changed with power oscillations from Fig. 5.4.a and 5.4.b. But it raised some concern about the cladding creep behavior which is above the content of current work.

We should be careful for the discussion above because we are taking the internal gas pressure as an input, which may be totally different with the real case. Thus, we discard the input for the internal gas pressure history and conducted analysis again. Its result was shown in the Fig. 5.5. The curve shape of the each parameter was basically the same, while the amplitude was a little different. This proves that the internal gas pressure only has very limited effects on fuel deformation in power operation conditions, when the coolant pressure is high enough.



(a) Plenum Pressure (b) Fuel Stack Axial Elongation (c) Clad Axial Elongation

(d)Effective Cladding Stress (e) Cladding Effective Strain (f) Gap Interface Pressure

Fig. 5.5 Results of FRAPTRAN simulation for BWR power oscillation (4), different internal gas pressure

It could be concluded that under a certain amount of undamped power oscillation cycles, the cladding would not fail and the fuel integrity is conserved.

6. Cladding Fatigue Analysis

Cladding fatigue failure is an unlikely failure mode for a reactor in base load operation, and it is omitted in the discussion in Section 5, as there are only 24 cycles in the oscillation calculation. But under hypothetical unstable power oscillations arising during ATWS, the oscillations could persist for thousands of times, given the short interval of 2 seconds between each cycle. Thus, we still need some consideration to the cladding fatigue issue in case of long-time power oscillations.

Upon ASME B&PV Code, cumulative number of strain cycles shall be less than the design fatigue lifetime with appropriate margins. The cumulative number of strain cycles shall be less than the design fatigue lifetime, with a safety factor of 2 on stress amplitude and a safety factor of 20 on the number of cycles. We were using ASME Code to conduct our cladding fatigue analysis.

In power oscillation conditions, the alternating stress is the thermal stress, $S_{alt} = 1/2\Delta\sigma_{\max,Tresca}$. From Fig.5.4.c and 5.4.d: $S_{alt} \le 20MPa$ under the oscillation conditions calculated in Section 5. By using Soderberg Criteria:

$$\frac{K_t\sigma_a}{S_N} + \frac{\sigma_m}{S_y} \le 1 \tag{6.1}$$

In which, $\sigma_a = 20MPa, \sigma_m = 400MPa, S_y = 500MPa, K_t = 2$, we can calculated $S_N = 200MPa$. From Fig. I-9.1 in *1998 ASME Boiler & Pressure Vessel Code III Devision 1*, the allowed number of cycles is greater than 10^5 under this alternating load, which means the time allowed for this oscillation is:

$$T_{total} = N * T_{period} = 10^5 \times 2 = 2 \times 10^5 \sec = 2.3day$$

The total allowed oscillation time indicate that the BWR doesn’t have cladding fatigue problems under power oscillations.

The small amplitude of alternating stress S_{alt} could be explained that the heat flux is proportional to the 3rd to 4th power of temperature difference between the wall and the coolant, cladding temperature changes are expected to be very small, unless dryout takes place. The average cladding temperature change in the power oscillations in Section 5 is shown here in Fig. 6.1. In the Fig., the average cladding temperature oscillation was only around 4ºC.

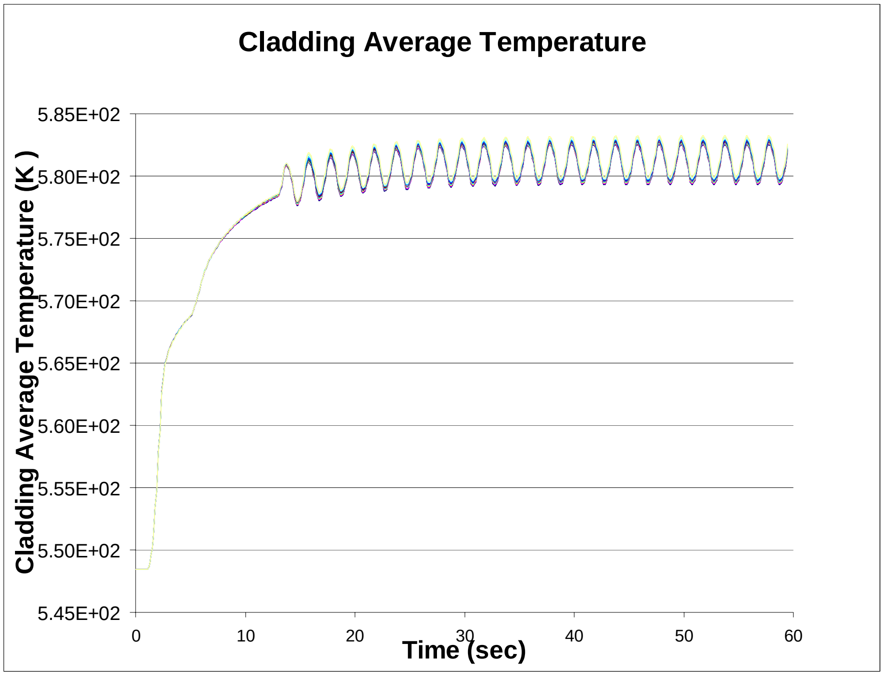


Fig. 6.1 Cladding Average Temperature Changes in Power Oscillations

7. Summary

In summary, the fuel integrity under power oscillations with ATWS was examined in this work by using FRAPTRAN code. Based on the analysis, the following conclusions could be drawn:

(1) FRAPTRAN code could be used to analyze fuel performance under BWR power oscillations. The FRAPTRAN code results were in good agreement with FK-11 test in Japan.

(2) The fuel deformation was mainly caused by PCMI and was roughly proportional to the fuel enthalpy. Enhanced cladding deformation due to ratcheting was not found.

(3) The cladding could satisfy the design criterion (stress-strain criterion) in ASME Code under power oscillation conditions, which means it could maintain the fuel integrity.

(4) The fission gas release (internal gas pressure) was of importance to tests in cold zero power conditions, but not the normal power operating conditions.

(5) Cladding thermal fatigue is not an issue under power oscillations, unless dryout takes place.

It should be pointed out there that the analysis in current study is based on consideration of stress-stain criterion and thermal fatigue. Cladding creep analysis, fission gas release model, boiling transition (dryout), and stress corrosion cracking are not considered in the analysis. These may weak the conclusion drawn above.

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