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An Assessment of Fuel Cladding Corrosion for Extended-Cycle Operation in PWRs

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Technical Report

Assessment of PWR Fuel Cladding Corrosion for Extended-Cycle Operation

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Prepared by
Korea Electric Power Research Institute
103-16, Munji-Dong
Yusung-Gu, Taejon
305-380 Korea

Author
Moon-Ghu Park

Prepared for
Electric Power Research Institute
3412 Hillview Avenue
Palo Alto, California 94304

EPRI Project Manager
Bo Cheng
Light Water Reactor Fuel Target

Nuclear Power Group

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REPORT SUMMARY

As the nuclear industry extends the fuel cycle length, waterside corrosion of Zircaloy cladding has become a limiting factor in PWR fuel design. This report presents an assessment of the effects of primary water chemistry, fuel thermal-hydraulic parameters, and type of cladding material on Zircaloy-4 cladding corrosion under extended-cycle operation in PWRs. The results indicate that with optimized Zircaloy-4 cladding, a rod average burnup of 50-55 GWd/MTU can be achieved without exceeding the prevailing 100 μm oxide limit.

Background

Zircaloy-4 fuel cladding has demonstrated an excellent performance record in PWRs. However, as the fuel duty and burnup increase, cladding corrosion can reach ~100 μm, beyond which oxide spalling and excessive hydrogen uptake by the cladding have been reported. Many plant chemistry factors such as higher lithium/boron concentration in the primary coolant can influence the corrosion behavior of Zircaloy cladding. The chemistry effect can be amplified in higher duty fuel, particularly when surface boiling occurs. An analysis of the impact of fuel design and plant operational parameters on cladding corrosion for high duty and extended burnup is therefore beneficial in preventing plants from reaching the undesired high corrosion level.

Objectives

To assess the effects of various fuel design and plant operational parameters on fuel cladding corrosion for extended-cycle operation in PWRs.

Approach

Investigators applied EPRI s PWR Fuel Cladding Corrosion (PFCC) model to fuel design and plant operational data from Korean PWRs, assessing the impact of various parameters on fuel cladding corrosion to a rod average burnup of 60 GWd/MTU. These parameters included reactor water chemistry and operating variables, with emphasis on lithium concentration, cladding surface temperatures, fuel rod power peaking history, core inlet temperature, mass flow rate, and cladding alloy tin composition. They evaluated two cases: Vantage-5 fuel design in a standard

Westinghouse plant and Combustion Engineering (CE) fuel design in a typical CE plant.

Results

Results of this study indicate that Zircaloy-4 cladding does not have sufficient margin in terms of corrosion and hydriding beyond a burnup range of 50-55 GWd/MTU. However, the actual burnup limit may be influenced by the thermal-hydraulic history of the fuel. Through optimization of the core design, excessive cladding corrosion can be avoided at the burnup range of 50-55 GWd/MTU. EPRI's PFCC model has proven to be a useful tool for optimizing the core design and managing fuel cladding corrosion.

This report discusses the separate effects of the following variables which have been known to contribute to in-reactor corrosion of Zircaloy cladding under extended-cycle operation: lithium concentration, tin content of cladding, power peaking factor, core inlet temperature, and mass flow rate. The report also considers two types of lithium chemistry regimes: extended-cycle chemistry schemes with initial lithium concentrations of 3-3.5 ppm and elevated chemistry schemes with initial lithium concentrations of 2.5-3.5 ppm. For an extended chemistry regime, the research indicates that lithium concentration has only a small effect on cladding corrosion. The report concludes with cladding corrosion predictions for Westinghouse and CE fuels.

EPRI Perspective

This study successfully used EPRI's PFCC model to evaluate the effects of reactor chemistry and operating variables on Zircaloy cladding corrosion for extended-cycle operation of Westinghouse and CE plants. Based on the results of the model calculations, special issues associated with cladding corrosion should be considered for extending the cycle length of PWRs beyond 18 months. These issues include substantial subcooled boiling as well as lithium concentrations that exceed the current experience base of up to 3.5 ppm. In addition, for burnup exceeding 50-55 GWd/MTU, the low-tin Zircaloy-4 cladding offers little corrosion margin, and a higher corrosion-resistant cladding alloy will be needed. This study will help utilities better understand such issues while meeting the challenges associated with PWR extended-cycle operation and managing the effects of chemistry changes on fuel performance and reliability.

TR-109047

Interest Category: Fuel assembly reliability and performance

Keywords: PWR, PWR fuel cladding performance

ABSTRACT

As the nuclear industry extends the fuel cycle length, waterside corrosion of Zircaloy cladding has become a limiting factor in PWR fuel design. Many plant chemistry factors such as higher lithium/boron concentration in the primary coolant can influence the corrosion behavior of Zircaloy cladding. The chemistry effect can be amplified in higher duty fuel, particularly, when surface boiling occurs.

In this study, the effect of reactor chemistry and operating variables on Zircaloy cladding corrosion was investigated and simulation studies using EPRI PWR Fuel Cladding Corrosion (PFCC) model were performed to evaluate the optimal primary chemistry condition for extended cycle operations. Other parameters, such as fuel rod power peaking factor and cladding alloy tin composition were also evaluated.

ACKNOWLEDGMENTS

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INTRODUCTION

In light water power reactors, like pressurized water reactors (PWRs) and boiling water reactors (BWRs), zirconium based alloys have been used as the fuel cladding and structural components material, because of their low neutron absorption cross-sections. Zircaloy-2, an alloy with ~1.5% Sn and 0.3% Fe+Cr+Ni, and Zircaloy-4, composition similar to Zircaloy-2 but free of Ni, have been used in BWRs and PWRs, respectively. In the USSR, a binary Zr-Nb alloy with 1% Nb has been selected for the Soviet PWRs. The behavior of these alloys as fuel claddings and other components has been satisfactory over the past decades. However, with increasing fuel duty and burnup, the good corrosion performance record of the Zircaloys has been challenged in many cases.

Fuel failures attributable to primary water chemistry conditions have occurred only in isolated cases in the past two decades. In BWRs, copper[1] and sodium[2] in reactor water have been implicated. In PWRs, only one water chemistry related fuel failure has been reported since the industry adopted reactor coolant system (RCS) pH control using LiOH [3]. Besides fuel failures, crud deposition on PWRs fuel have been reported to cause local power suppression resulting in axial offset anomalies[4].

The Zircaloy-4 fuel rod cladding forms a barrier against the release of the radioactive fission products formed during operation. Maintaining cladding integrity is a major objective of the plant operator and licensing authorities.

At normal operating temperatures, the Zircaloy corrosion reaction can be described by:

$$Zr + 2H_2O \rightarrow ZrO_2 + 2H_2 \qquad \text{(eq. 1-1)}$$

The major effect of corrosion on cladding integrity results from two related processes: 1) the formation of a ZrO_2 layer causing cladding thinning and 2) the hydriding of Zircaloy causing cladding embrittlement. Both phenomena degrade cladding integrity. The rate of Zircaloy corrosion is controlled by the temperature of the metal-metal oxide (Zr-ZrO_2) interface. Since the cladding transfers heat to the coolant, the interface temperature during operation is always higher than that of the coolant. As the oxide grows thicker, the interface temperature increases, which in turn increases the corrosion rate. This effect is the major factor leading to accelerated corrosion at high burnups.

Additives have been added to the primary chemistries in recent years in both PWRs and BWRs to protect aging plant system materials against stress corrosion cracking (SCC) or reduce shutdown radiation exposure. The chemistry changes have taken place concurrent with a trend of increasing fuel thermal duty, thus increasing the propensity of fuel to interact with the additives. Thus, the effects of chemistry changes on the corrosion and hydriding behavior of Zircaloy fuel cladding and deposition of crud on fuel rods have been evaluated in several fuel surveillance programs under the sponsorship of EPRI and host utilities[4,5,6,7]. For PWRs, in-core loop tests were performed to evaluate the effects of elevated lithium and zinc injection on fuel rod corrosion and hydriding. The results have formed the bases for inputs to the recent versions of EPRI Primary Water Chemistry Guidelines[8,9].

Recently, extended cycle length operation of PWRs has become a major issue in nuclear industry. To comply with the increase in cycle length, it also requires increases in the concentrations of boric acid and lithium hydroxide. The effects of these chemistry changes on fuel performance and reliability are subjects of increased interest.

1.1 Lithium Effect on Cladding Corrosion

Uniform corrosion of Zircaloy-4 cladding has become a major concern for high burnup fuel design. The oxide thickness of low tin Zircaloy-4 cladding has been reported to reach ~100μm at a rod burnup of 50-60 GWd/MTU[10,21]. As the thermal duty of fuel rods increases due to higher energy core design, the cladding corrosion rate is expected to further increase. The decreasing corrosion margin in high duty and high burnup fuel design has imposed a constraint on changing the RCS chemistry for other purposes, such as increasing pH (and Li) for radiation dose rate control and injecting Zn for mitigation of Inconel stress corrosion cracking. Recent improvements in Zircaloy cladding material have resulted in 10-30% reduction in the oxide thickness, as claimed by the vendors. However, such corrosion improvements have not yet yielded enough margin for further burnup extension or for tolerating further increases in thermal or chemical duties during operation. One example of higher chemical duties for Zircaloy cladding is higher Li/pH in the primary coolant.

The conditions that have existed in the past, namely lithium at <2.2 ppm and pH(300°C) at ≥6.9, cannot be easily attained when the cycle length is stretched from 12 to 18 or 24 months and the startup boron concentration is increased from ~1200 ppm to 1400-1600 or 1700 to 2000 ppm, respectively (Actual concentration depends on the enrichment and amount of burnable poisons in the fuel design.). In the early 1980 s, most plants adopted Coordinated Chemistry (Figure 1-1) in which lithium is coordinated with boron to maintain a pH of 6.9. In late 1980, Elevated lithium Chemistry (Figure 1-2) regime was tested in four US PWRs to reduce nickel ferrite precipitation on fuel and thus reduce plant radiation levels. Solubility studies suggest that operating in the pH=7.2-7.4 region will minimize nickel ferrite precipitation on lower core surfaces and prevent precipitation on upper core areas. This scheme was terminated after one cycle due to concerns over the potential effects of prolonged exposure to 3.5 ppm lithium on primary water stress corrosion cracking (PWSCC) of Alloy 600 and Zircaloy cladding corrosion. This results in the Modified Chemistry scheme currently adopted by most US PWRs[9], as shown schematically in Figure 1-3. The modified chemistry results in an increase in the integrated time of exposure of fuel to lithium in the coolant. Maintaining lithium concentrations <2.2 ppm during startup, however, can result in a pH(t)<6.9 for plants operating with an initial criticality boron concentration exceeding ~1200 ppm. An Extended Cycle Chemistry regime as shown in Figure 1-4 has been under discussion. In this regime, the startup lithium and boron are coordinated to maintain pH(t) ≥6.9 until the required lithium concentration decreases to 2.2 ppm. At this point, the pH(t) is permitted to increase with a constant 2.2 ppm Li, similar to the Modified Chemistry regime. A final pH(t) =7.2-7.4 is implemented. It is important to note that implementing an Extended Cycle Chemistry regime may require significant concentrations of lithium for the early portion of the cycle, as shown in Figure 1-4. Specifically, a startup lithium value of ≥3.5 ppm may be required for a 20-24 month cycle core. For a 24-month cycle, a startup lithium of ~3.5 ppm may be needed to keep the pH(300°C) at 6.9. Thus Extended Cycle Chemistry regime is under consideration. As can be seen in Figures 1-1 to 1-4, the integrated exposure of fuel to lithium throughout one fuel cycle is increased significantly when the chemistry is changed to Elevated Chemistry. An intermediate increase can also be seen in the case of Modified or Extended Cycle chemistry. To evaluate the potential effects of the increased exposure to lithium on fuel cladding corrosion, fuel surveillance programs were implemented to obtain cladding oxide thickness data when the Elevated Chemistry regime was implemented at four plants in the late 1980s. Results from three of the four plants (St. Lucie-1, Oconee-2, and Millstone-3) have been analyzed by fuel vendors [6,13,14,15]. In all three cases, the Elevated lithium chemistry has been attributed to increase the oxide thickness in the order of 10-15%. In each case, conclusions were made with ambiguity because of large scatter in the data. An additional analysis of the Millstone-3 EOC3 data by EPRI/Nuclear Electric, however, suggests that the apparent higher oxide data cannot be attributed to the Elevated Chemistry exposure[6]. Because of the ambiguity, additional oxide data were obtained recently at Millstone-3 EOC4 to further assess the effects. Preliminary conclusions by Westinghouse indicate that no significant oxide thickness increase can be detected on some reload fuel rods exposed to the elevated chemistry.[16] Apparently, no consensus on the extent of the effect of Li/pH on fuel cladding corrosion in PWRs has been reached.

However, the validity of the predicted effects of various lithium chemistry regimes on cladding corrosion should be further verified for the following conditions: (1) oxide thickness exceeding 80-100μm when cracking of the oxide may occur, (2) significant boiling, and (3) crud deposition. In particular, crud deposition may cause hideout of Li, leading to significant corrosion enhancement. This is because lithium hydroxide is among a few chemical species that can accelerate Zircaloy uniform corrosion in 300-400°C water. Ex-reactor coupon tests[17] and heated rod tests[18] have shown that Zircaloy corrosion enhancement due to lithium at <10 ppm is small, particularly in presence of B>100 ppm, when the Zircaloy surface has no or low void fraction (<5%). At higher lithium concentrations and high void fractions, corrosion enhancement was detected. Thus, for high duty fuel with significant surface boiling, an assessment of the impact of lithium on cladding corrosion should be made.

EPRI has undertaken a parametric test program to study the effect of primary coolant Li/B/pH on fuel cladding corrosion. The program was funded by the NFIR group and performed by French CEA-Cadarache Laboratory[18,19]. The results from the test program have been incorporated into a new EPRI Fuel Rod Corrosion (EPRI/PFCC) Code after a calibration is made using fuel cladding oxide data[20,21,22]. The effects of Li/pH on corrosion of Zircaloy cladding operated under the various chemistry regimes are then assessed using the PFCC Code.

1.2 Temperature and Boiling Effects on Cladding Corrosion

As the fuel duty is increased, the hot channels in PWR cores will experience more severe thermal-hydraulic conditions. The increase in thermal duty results in the increase in rod power, and cladding temperature which has led to several studies on the effect of local boiling [5,19]. It is known that an increase in the fuel cladding corrosion rate can occur under subcooled boiling conditions. One postulated explanation is that significant subcooled boiling leads to local radiolytic production of oxidizing species, which can increase the corrosion rate of Zircaloy. Another possibility is the formation of crud deposits during operation to cause a thermal impedance effect or hideout of lithium inside the crud deposit. A semi-qualitative method has been developed to evaluate the propensity for boiling in PWRs[5]. However, there is insufficient data available to accurately determine the exact void fraction or power peaking which leads to accelerate cladding corrosion.

1.3 Crud Deposition Issues and Chemistry Effects

Crud deposition on fuel rods has been reported in PWRs prior to implementation of Li/pH control[7,23]. Core pressure drop, core radial power offset and enhanced cladding corrosion were reported in a few cases. Following introduction of the Li/pH control, no crud related fuel performance issue was reported during the 1980s. In recent years, axial and radial power offsets[3,4] and localized cladding corrosion [24] have been reported to result from deposition of crud on fuel rod surfaces. Subcooled boiling is suspected to be related to the crud deposition in those cases.

1.4 Silica Issues

Many PWRs have recently experienced high silica concentration in the RCS exceeding the 1 ppm diagnostic limit recommended in the PWR Chemistry Guidelines[9]. The high silica concentration is mainly due to intrusion from the storage pool during refueling outages where degradation of Boraflex storage racks occurs. Silica accumulation due to reclamation of boric acid can also contribute to high RCS silica concentration. Data on effects of silica on Zircaloy corrosion and fuel rod crud deposition have become available only recently. In the mini-autoclave test at the Halden reactor, Zircaloy coupon samples were exposed to water containing 200 ppm silica in two separate experiments for ~120 days. No effect of the 200 ppm silica on Zircaloy corrosion and hydriding was detected [2].

1.5 Zinc Injection

Another potential chemistry change is addition of zinc to the RCS as a means to mitigate SCC of Inconel components, such as vessel head penetration and steam generator tubes. An evaluation of Farley-1 and -2 cladding oxide data was performed using PFCC model[11,22]. The results suggest that the most probable causes of the slightly higher oxide thickness data are: (l) higher fuel duty of some fuel rods when compared to previous cycles; (2) measurement interference due to the presence of the black deposit; (3) inherent variability of the cladding corrosion process. In addition to the plant demonstration at Farley-2, an in-core loop test was carried out at the Halden reactor to provide an early warning on any deleterious effects of zinc injection. The results showed no apparent effects of Zn on the corrosion and hydriding of the low tin and standard Zircaloy cladding [12].

1.6 Objective of This Study

This report represents a study on the effects of chemistry changes and reactor operating variables on PWR fuel cladding corrosion using EPRI PFCC (PWR Fuel Cladding Corrosion) code. In Section 2, the plant and fuel types introduced in this study and the corresponding PFCC inputs are explained. Cladding corrosion predictions are given in Section 3 for Westinghouse and CE type fuels. The effects of lithium concentration, Sn content, power peaking factors and cladding surface temperatures are considered.

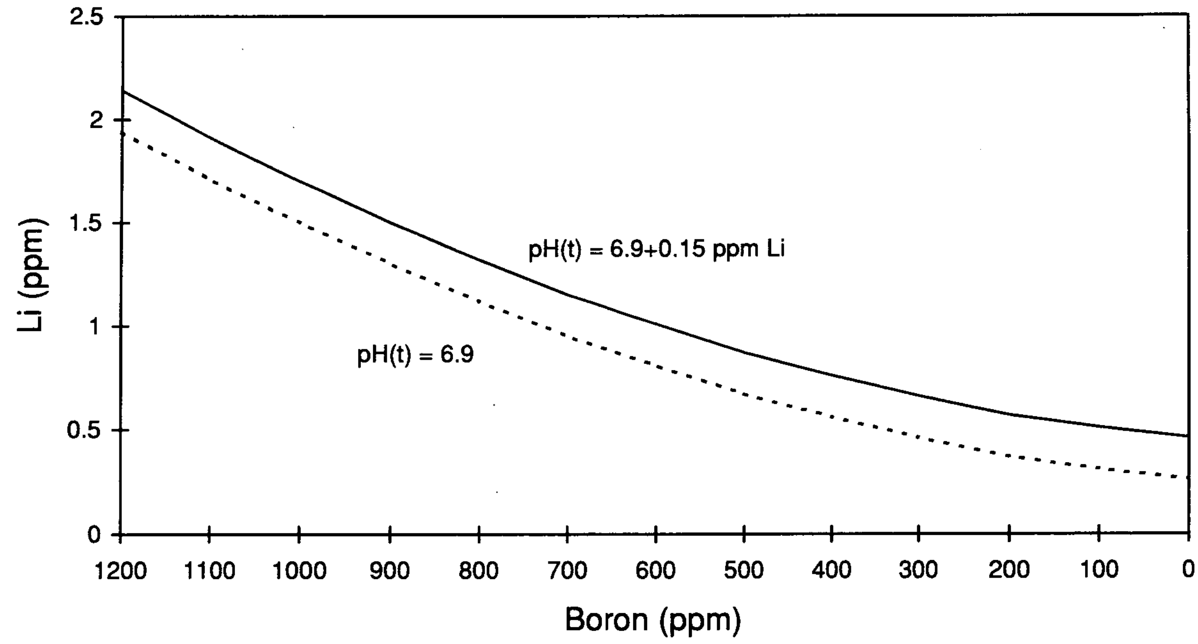


Figure 1-1
Coordinated Chemistry scheme

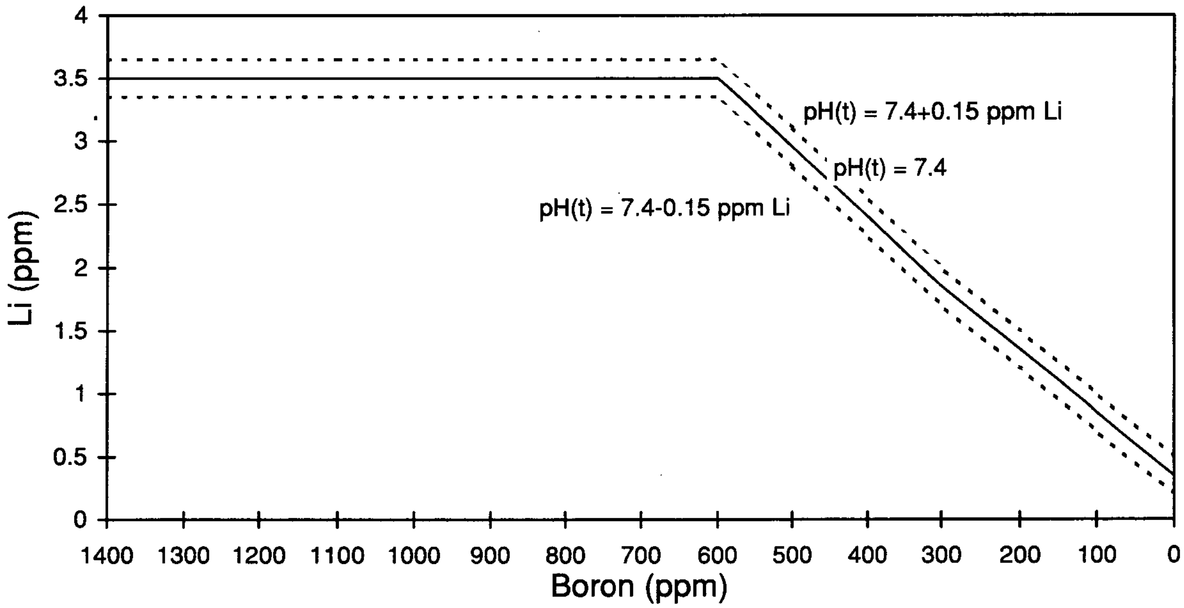


Figure 1-2
Elevated Chemistry scheme

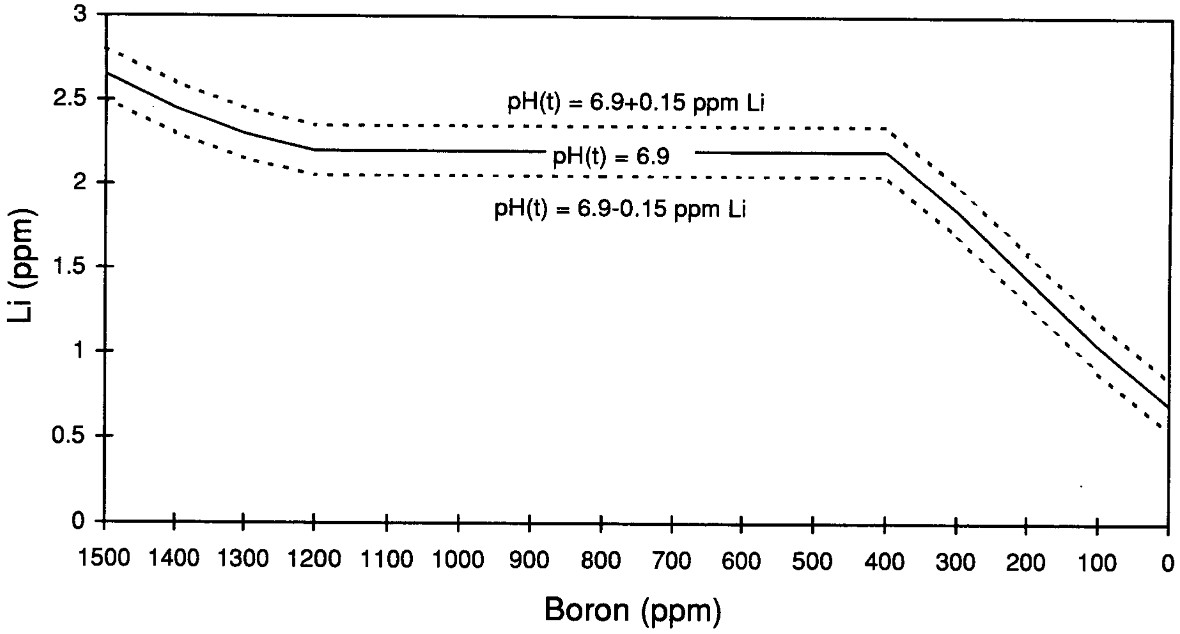


Figure 1-3
Modified Chemistry scheme

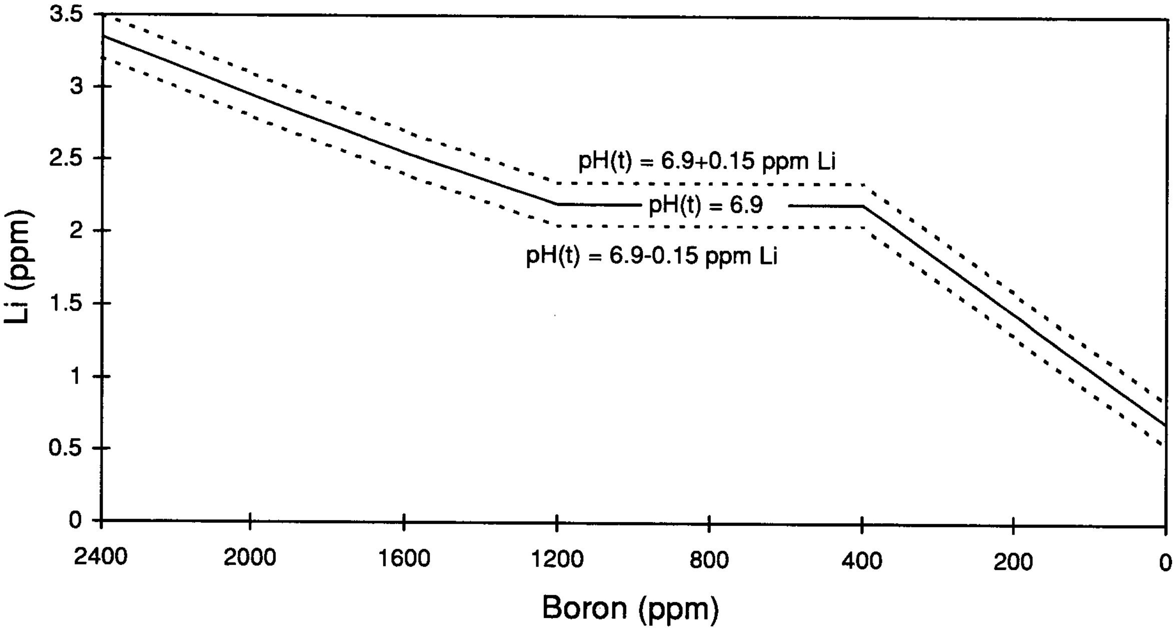


Figure 1-4
Extended Cycle Chemistry scheme

2

PFCC INPUT DATA REVIEW

Two types of reactor cores (plant A and B) with Korean produced fuel were investigated in the simulation. Plant A is a 970 MWe Westinghouse plant loaded with 17x17 Vantage-5 assemblies and plant B is a 1000 MWe CE plant with 16x16 fuel assemblies. The oxide thickness prediction calculations were performed using EPRI PFCC code. The methodology of the PFCC code can be found in EPRI report TR-105387V1,V2[20,21]. The required input data for the PFCC code includes oxidation model parameters, fuel geometry data, thermal-hydraulic data, system parameters, and operating history data. The PFCC input data for plant A and B are summarized in Appendix A (Table A1).

2.1 Oxidation Model Parameters

The oxidation model parameters include:

- Oxidation model selector

  PFCC has several kinds of oxidation predictor models (EPRI/SLI, EPRI/KWU/CE, NE PLC, COCHISE). The basic PWR formulation model (EPRI/SLI) was used.

- Oxide thermal conductivity data

  PFCC code has a capability of defining oxide thermal conductivity table as a function of oxide thickness. The default value for EPRI/SLI model (0.015 W/cm-K) was used.

- Pre- and post-transition model coefficients

The Zircaloy oxidation rate equations used are:

$$\frac{d(s^3)}{dt} = k_1 \exp(-Q_1 / RT_i) \text{ pre-transition regime } (s \le s_{trans}) \qquad \text{(eq. 2-1)}$$

$$\frac{ds}{dt} = k_2 \exp(-Q_2 / RT_i) \text{ post - transition regime (s > strans)} \qquad \text{(eq. 2-2)}$$

Where s = oxide thickness (μm)
R = ideal gas constant (cal/mol-K)
T_i = oxide to metal interface temperature
Q/R = activation energy (K)

The corrosion enhancement factors (k_1 and k_2) are the functions of empirical coefficients. These coefficients can be found in Appendix A.

- Metallurgical Variables

  The important metallurgical factors affecting the corrosion behavior are found to be: tin content, precipitate size and distribution, thermomechanical processing variables, carbon and silicon content. The effects of tin content, precipitate size and distribution, thermomechanical processing variables have been broadly characterized and incorporated into PFCC model. But, since the effect of carbon and silicon are most likely due to their effects on the distribution of Fe and Cr and the $Zr(Cr,Fe)_2$ precipitate, the PFCC code does not treat these elements separately. The size and distribution or the volume density of intermetallic precipitates have been shown to have strong effects on the nodular corrosion resistance of Zircaloy and on the uniform corrosion behavior. Figure 2-1 shows the intermetallic particle size distributions for both plants. Plants A and B are using low tin improved Zircaloy-4 cladding with 1.32%wt and 1.31%wt Sn, respectively.

2.2 Fuel Geometry Data

The fuel geometry data consist of the following:

- Fuel pellet and cladding geometry data
  The fuel type and geometry data are shown in Table 2-1.

- Hydraulic diameter and heated diameter

  The fuel design values of equivalent hydraulic diameter was used. The effective heated diameter is required to consider the variation of heated diameter from cycle to cycle [7]. In this study, constant heated diameter was assumed.

Table 2-1
Fuel Design Characteristics

Fuel Design	Plant A (W)	Plant B (CE)
Assembly geometry	17x17	16x16
No. of rods per assembly	264	236
Overall assembly length (mm)	4063	4528
Overall maximum assembly width (mm)	214	207
Rod length (mm)	3866	4528
Rod outside diameter (mm)	9.5	9.7
Clad material	Improved Zr-4	Improved Zr-4
Clad thickness (mm)	0.572	0.64

2.3 Thermal-hydraulic Data

The PFCC code uses the thermal-hydraulic methodology of the Single-Channel Model (SCM) primarily to support the cladding corrosion analysis work of PWRs. The code was streamlined for general use, and extension were added that would allow testing the impact of different corrosion models or the sensitivity of predictions to variations in the model parameters. The PFCC code predicts the cladding oxide buildup along the axial length of a single PWR fuel rod throughout exposure. It utilizes the energy balance in a single closed channel and heat transfer correlations to derive the thermal-hydraulic conditions of the coolant around a given fuel rod and the surface temperature of its cladding. The thermal-hydraulic data is composed of:

- Heat transfer coefficients

  Thom s correlation was used for the convective heat transfer coefficient.

- Flow and turbulent mixing factor

  The default values for non-uniform flow factor and turbulent mixing factor were used.

2.4 System Parameters

Nominal design values are used for system pressure and coolant inlet temperature. The thermal design value of the core mass flux was used instead of measured value which is normally 4~7% greater than thermal design value. These parameters data are shown in Table 2-2.

The system parameters specify:

- Reactor system pressure
- Inlet coolant mass flux
- Coolant inlet temperature
- Core average linear power density

Table 2-2
System parameters

Parameter	Plant A (W)	Plant B (CE)
System pressure (psia)	2250	2250
Coolant mass flux (kg/m^2-s)*	1.945E5	2.14E5
Inlet coolant temperature (°C)	291.4	295.8
Core average linear power density (kW/m)	17.83	17.69

* thermal design flow rate

2.5 Operating History Data

The operating history data block specifies:

- Power step time history information

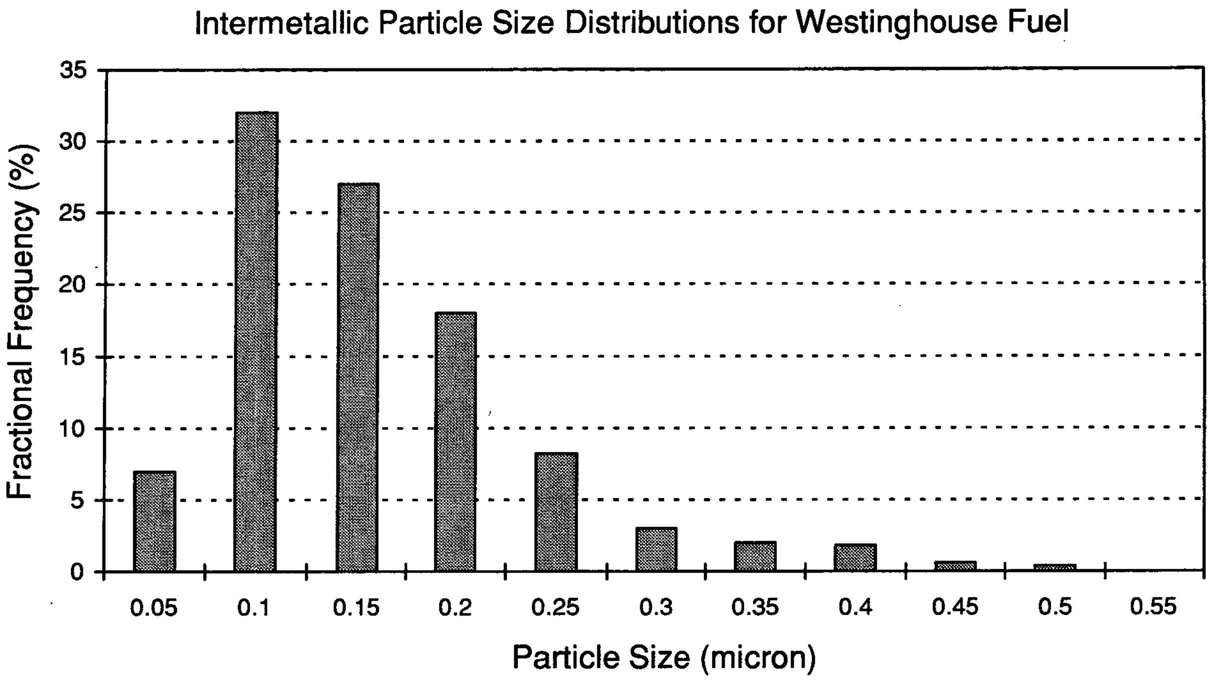
  The maximum RPF for each cycle of both plants are assumed to be 1.35, 0.8, 1.2, respectively (Figure 2-2). In the third cycle, the maximum peaking factor has been trending upward. This bounding feature is introduced to consider the loading of the twice burned fuel into the core center. However, this peaking factor history is compatible with the realistic power history not the bounding power history which might give too conservative results in cladding corrosion calculation.

- Axial power and flux shape profiles

  The three cycles of exposure history were simulated for plant A and B. The fast flux multipliers (FF Mult) are treated as constants. The axial power shape profile is assumed to have typical variation during a cycle operation of the reactor. Figures 2-3 to 2-5 show the variation of the axial power distributions for each plants.

- Lithium chemistry information

  The effect of lithium is simulated for extended cycle operation stretched from 12 to 18 or 24 months with Extended Chemistry and Elevated Lithium schemes.



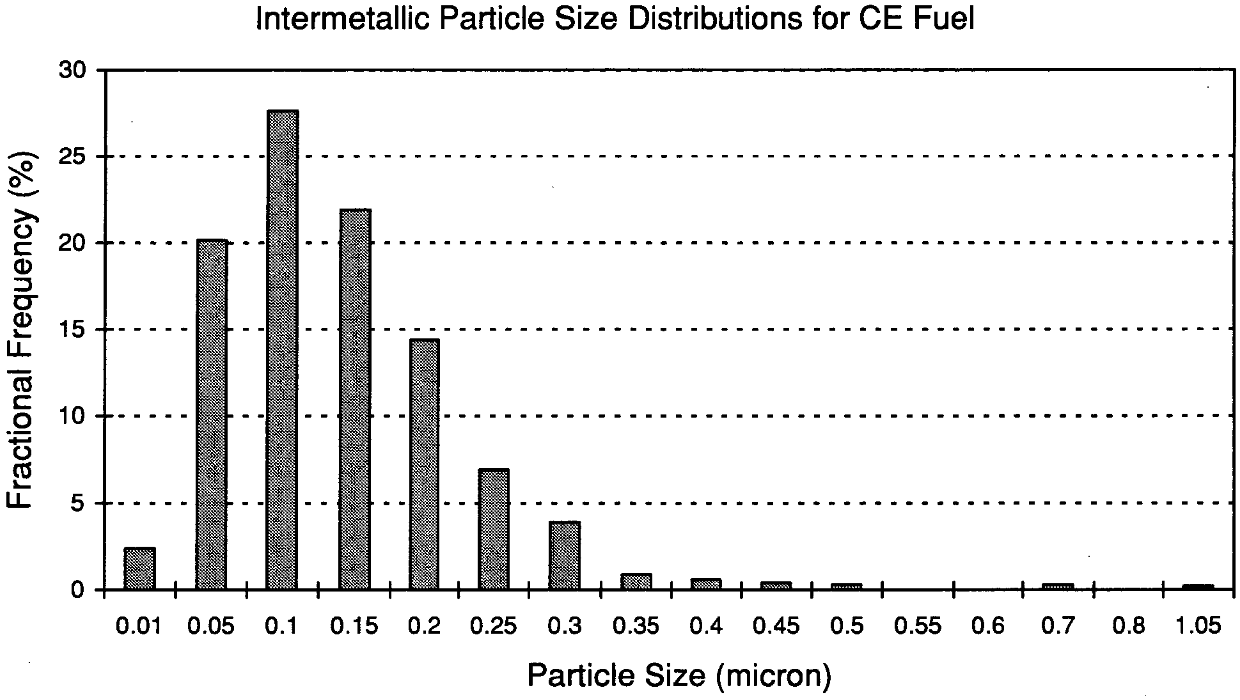


Figure 2-1
Intermetallic particle size distributions

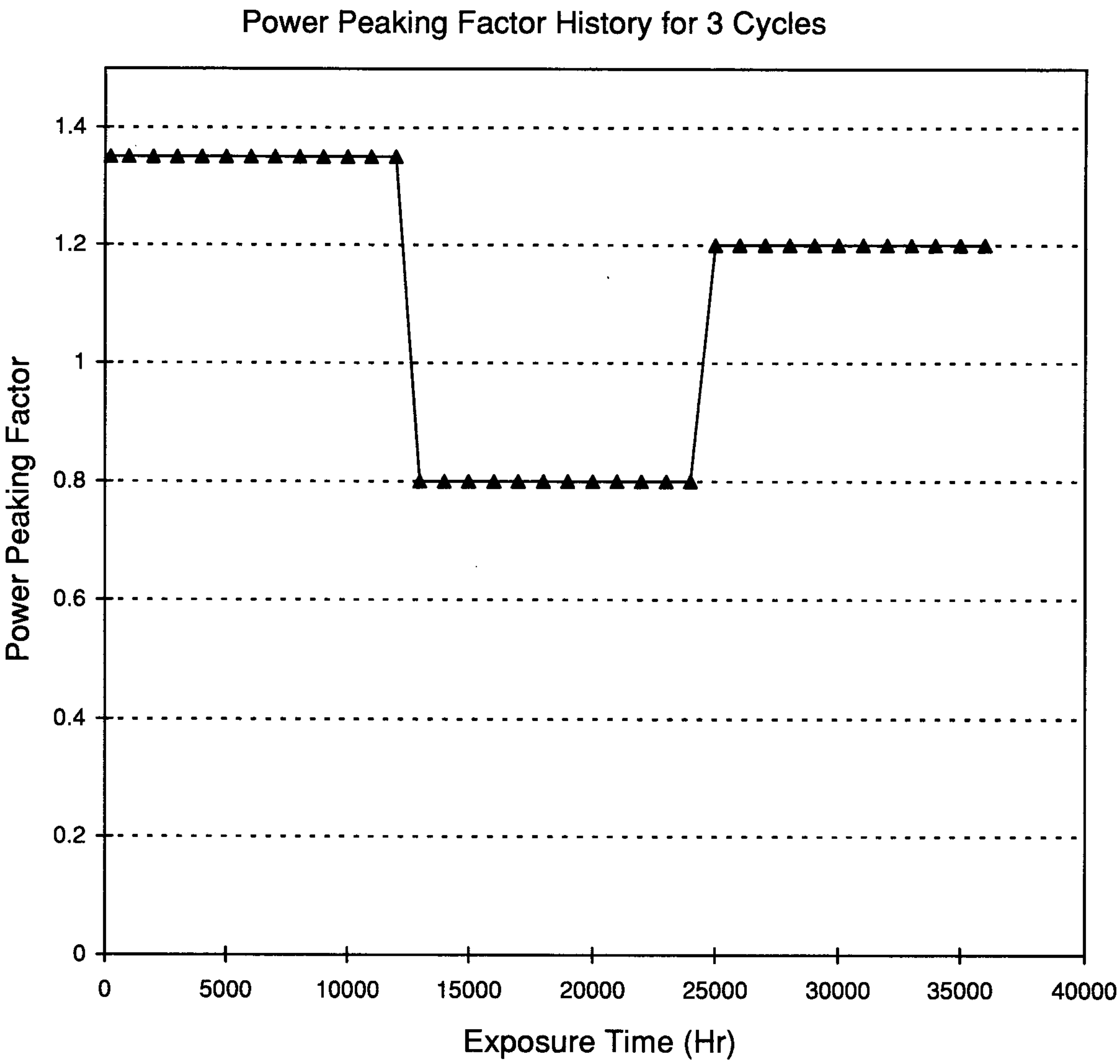


Figure 2-2
Power peaking factor history for 3 cycles

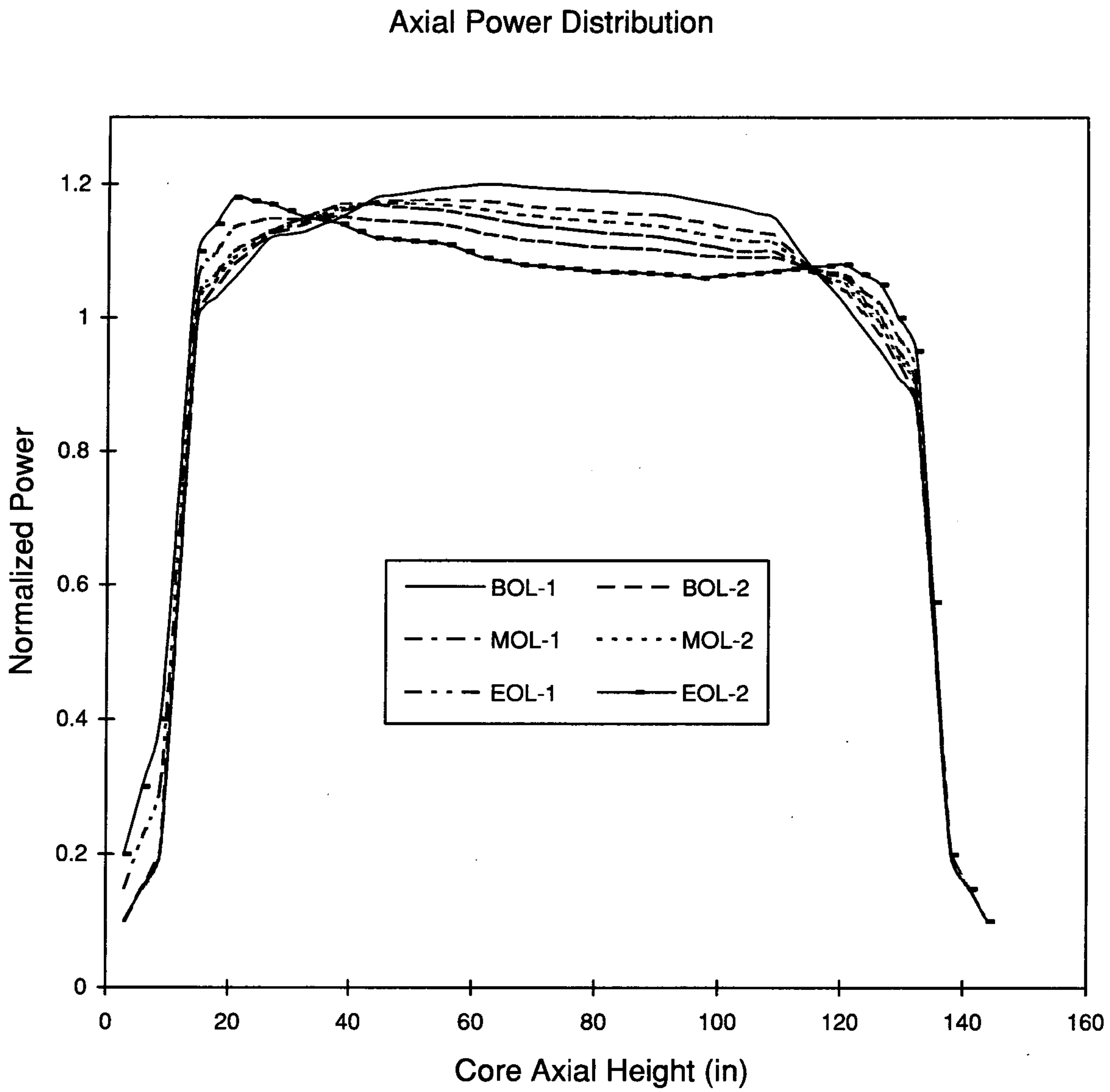


Figure 2-3
Axial power distribution for W Vantage-5 fuel with axial blanket

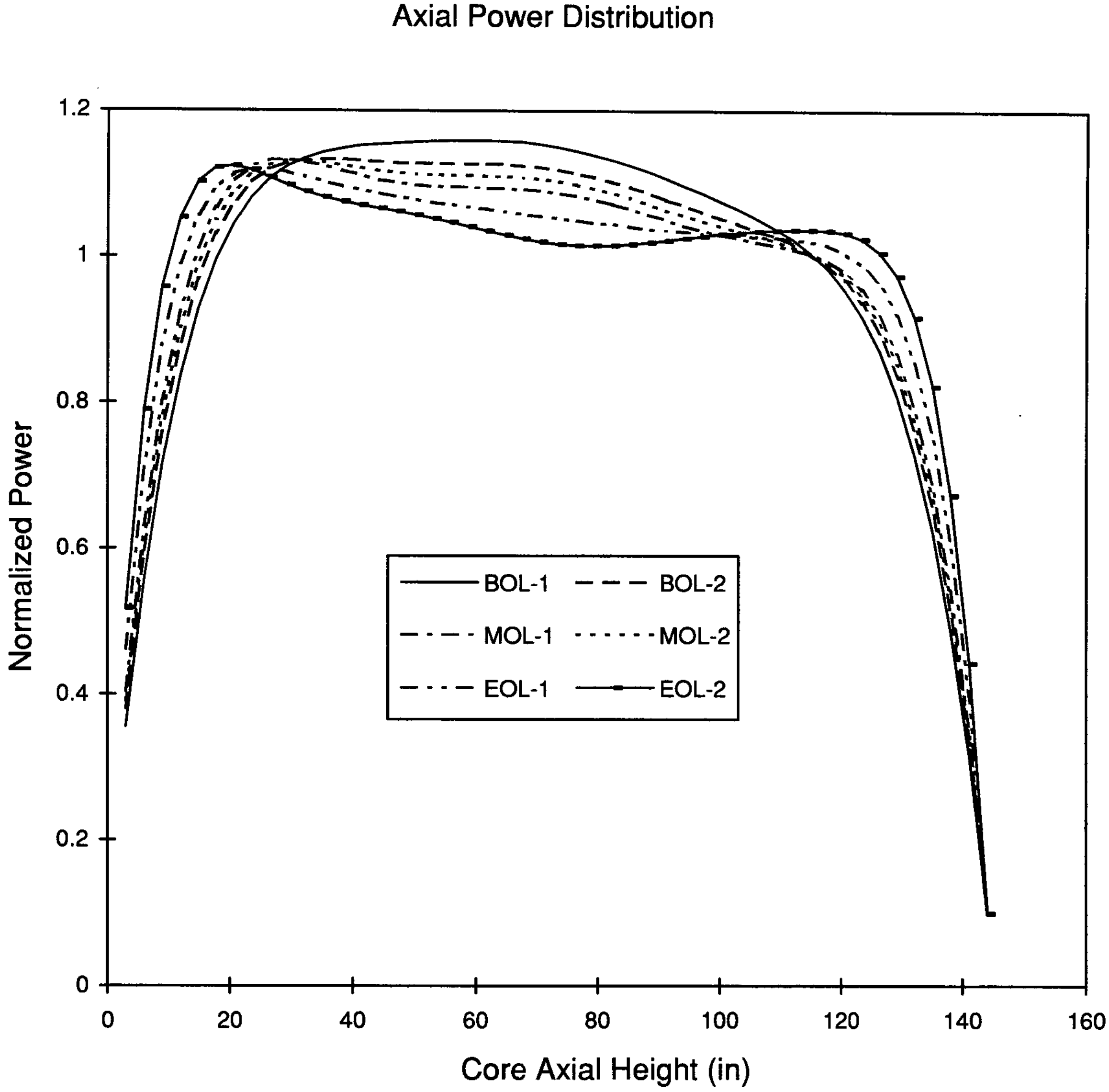


Figure 2-4
Axial power distribution for W fuel without axial blanket

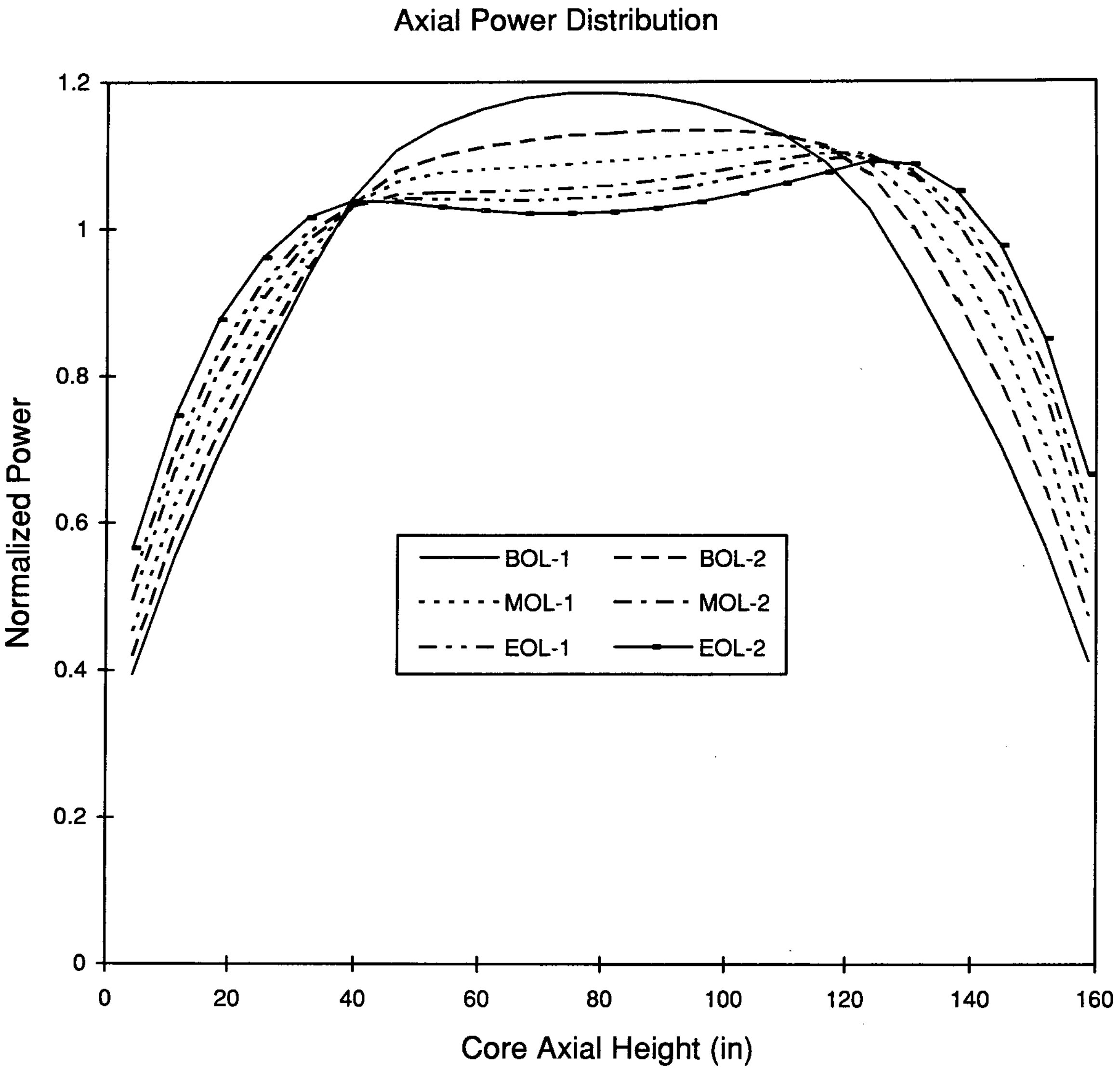


Figure 2-5
Axial power distribution for CE fuel

3

ANALYSIS OF PLANT CHEMISTRY EFFECTS ON FUEL CLADDING USING PFCC CODE

In this section, the separate effects of Zircaloy cladding corrosion under extended-cycle operation are discussed including the effects of lithium concentration, tin (Sn) content of cladding, power peaking factor and local boiling, etc. These factors have been known to contribute to in-reactor corrosion of Zircaloy cladding and are incorporated in the PFCC model which are shown schematically in Figure 3-1 [22].

3.1 Lithium Concentration

In the course of the PFCC development, the effect of pH coordinated lithium chemistry was a central focus for code application considerations. The benchmark and qualification study for the effect of lithium concentration was performed for ~3.5 ppm lithium by using 45 rods irradiated for one to five cycles in five different PWRs [21]. For 18~24-month cycle, a startup lithium of ~3.5 ppm may be needed to keep the pH(300°C) at 6.9. The oxide thickness of low tin Zircaloy-4 cladding has been reported to reach ~100 μm at a rod burnup of 50-60 GWd/MTU [10,20]. In this study, two types of lithium chemistry schemes are considered: Extended Cycle and Elevated Chemistry schemes with various levels of lithium concentration (Figure 3-2). The initial lithium concentrations of 3~3.5 ppm for extended chemistry scheme and 2.5~3.5 ppm for elevated chemistry scheme are considered. Although the 3.5 ppm elevated lithium program was terminated, the high lithium elevated scheme is simulated to check the effect of various lithium chemistry regimes on cladding corrosion. This scheme exhibited considerable increase in oxide thickness in extended-cycle operation which is due to the more integrated exposure to the high lithium concentration. The simulation results for plant A and B show no big difference in peak oxide thickness. But Figures 3-3 and 3-4 show ~13% increase in the peak oxide thickness when adopting elevated chemistry regime. This may be not a significant increase if there remains a enough corrosion margin, i.e., up to the burnup of 40~50 GWD/MTU. However, for extended-cycle operation, elevated chemistry regime with 3.5 ppm lithium gives 120~135 μm of oxide thickness at the burnup of 60 GWD/MTU. Extended Chemistry scheme also gives 115~120 μm of oxide thickness at the burnup of 60 GWD/MTU. For high burnup fuels with the possibility of oxide cracking, which may occur at oxide thickness exceeding 80-100μm, special concerns should be implemented to assure the integrity of

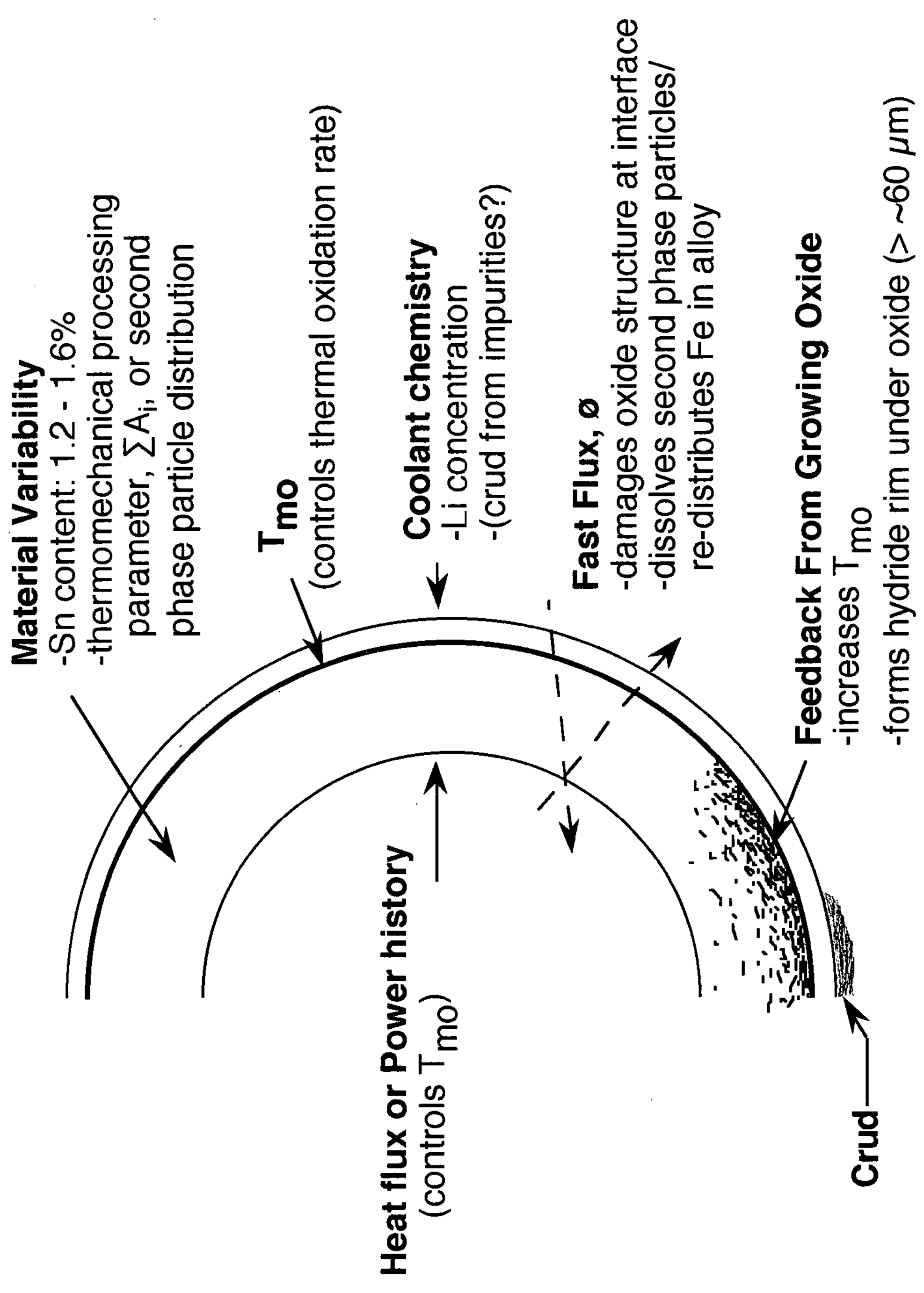


Figure 3-1
Factors known to contribute to Zircaloy cladding corrosion

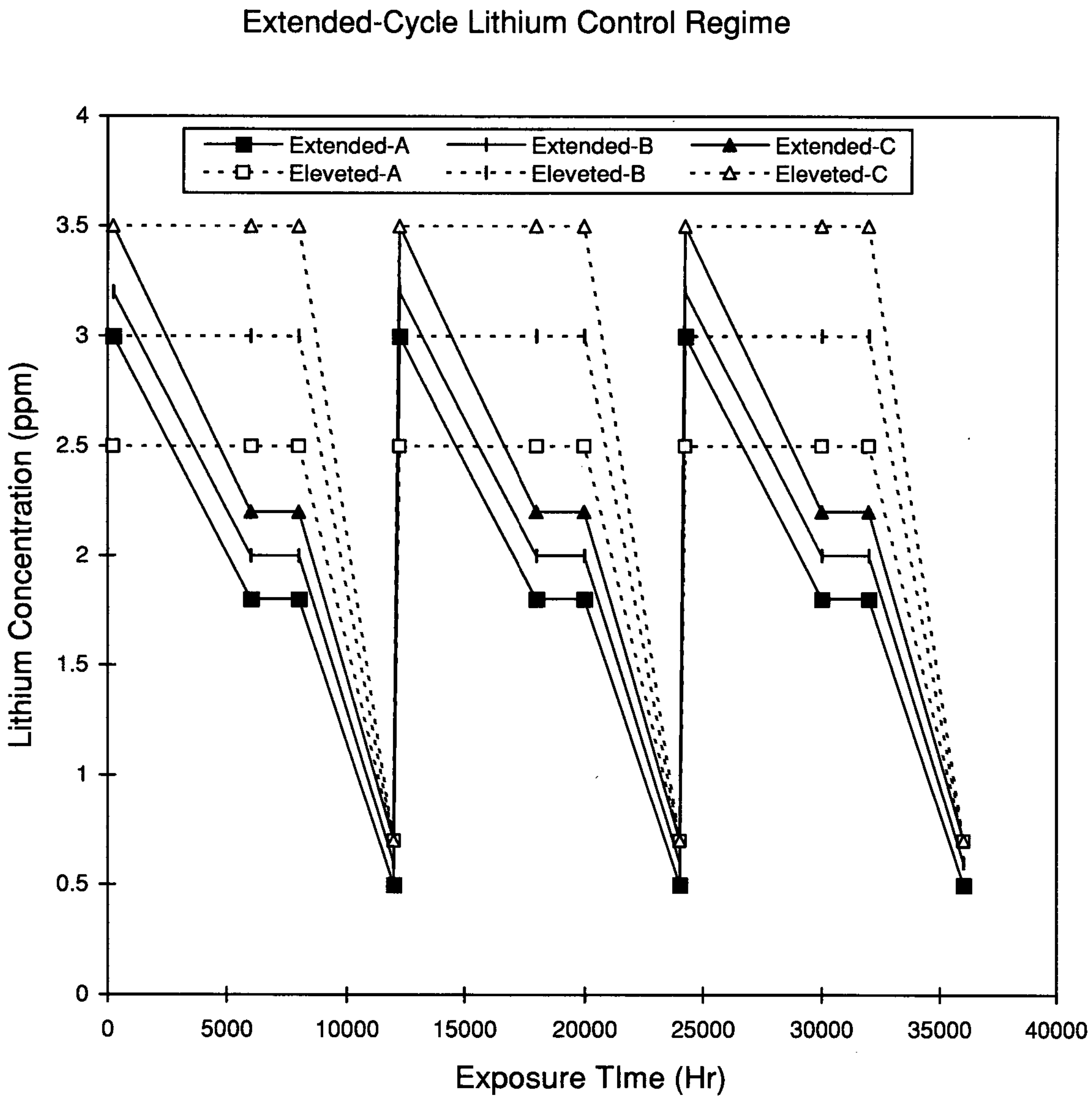


Figure 3-2
Lithium histories

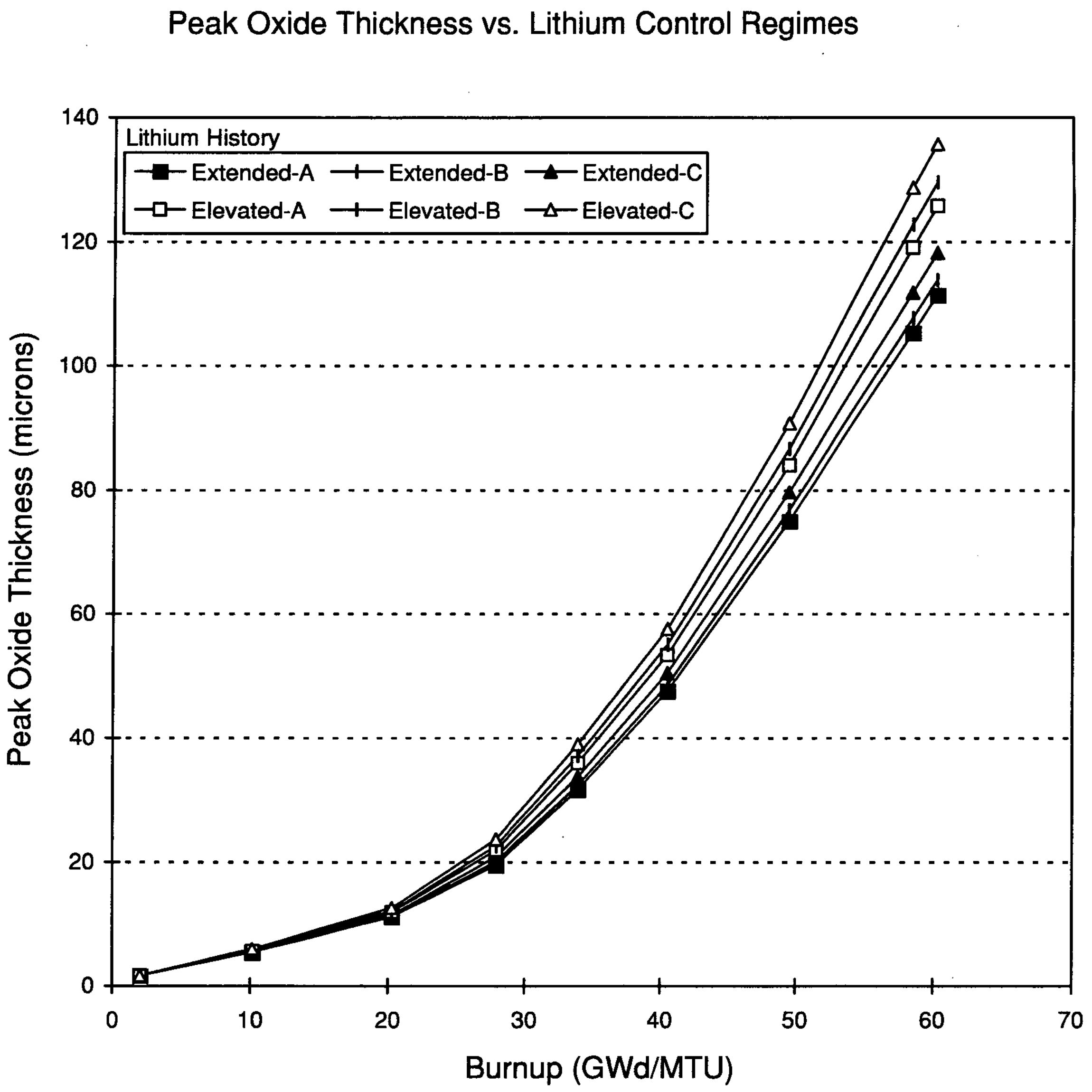


Figure 3-3
Peak oxide thickness vs. lithium control regimes for W fuel (Tin=291.4°C, Pavg=17.83 kW/m)

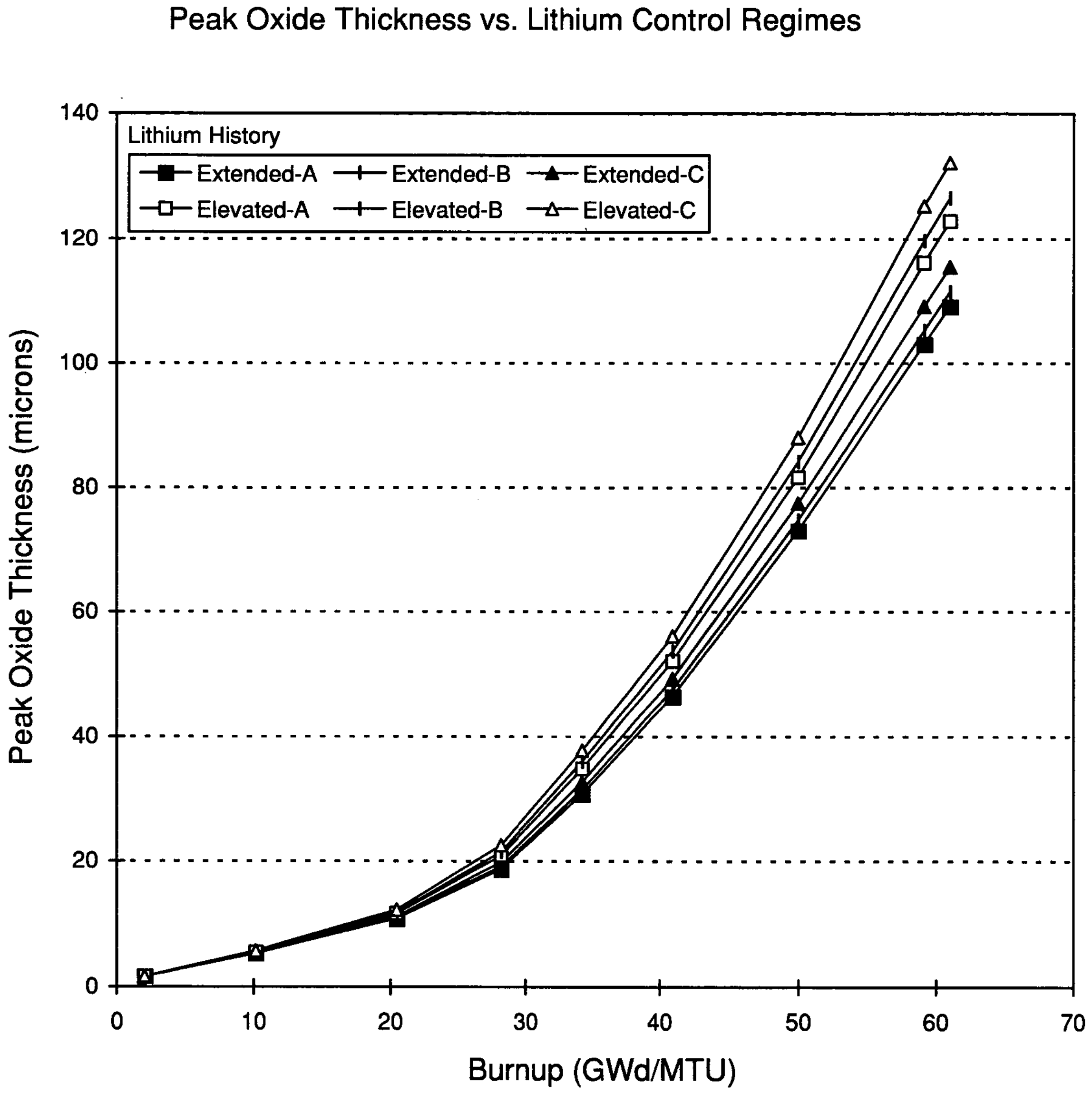


Figure 3-4
Peak oxide thickness vs. lithium control regimes for CE fuel (Tin=295.3°C, Pavg=17.69kW/m)

the cladding. As shown in Table 3-1, the sensitivity of lithium concentration for Extended Chemistry scheme is very low compared with that of Elevated Lithium Chemistry regime. Thus, for extended-cycle operation, Extended Chemistry scheme can be a viable approach to control the pH of RCS.

3.2 Tin (Sn) Content of Cladding

Ex-reactor Zircaloy corrosion data had suggested that reducing tin content might improve fuel cladding corrosion resistance. In NFIR-I, a series of projects involving several pre-characterized Zircaloy-4 variants, autoclave tests, out-reactor loop tests, and a lead test assembly program in a PWR produced convincing evidence of the superior corrosion resistance of low-tin Zircaloy as compared with standard Zircaloy. All major vendors have since changed their Zircaloy specifications to low-tin Zircaloy-4 [25]. The more recent results in References ([26,27,28]) have suggested that improved low tin cladding can reduce the oxide thickness at high burnups by about 20-40%. Lowering tin content is believed to reduce the solubility of Fe, hence reducing the uniform oxide growth rate. However, it has been noted that lowering tin alone is not sufficient to improve the corrosion resistance. Proper processing control of the Zircaloy cladding is needed to achieve the beneficial effect of lowering tin content [22]. The published data include a low tin content in the range of 1.2 to 1.4%wt. As noted previously, tin content has a very significant effect on cladding corrosion as shown in Figures 3-5 and 3-6. It reaches a thickness ~120 µm for the nominal tin contents of each type of fuels at a burnup of ~60 GWD/MTU. The measured nominal values and standard deviations of tin content of plant A and B are 1.32%wt, 0.04%wt and 1.31%wt, 0.02%wt respectively. As shown in Table 3-1, there seems no big difference in the sensitivity factors of tin content for each plant.

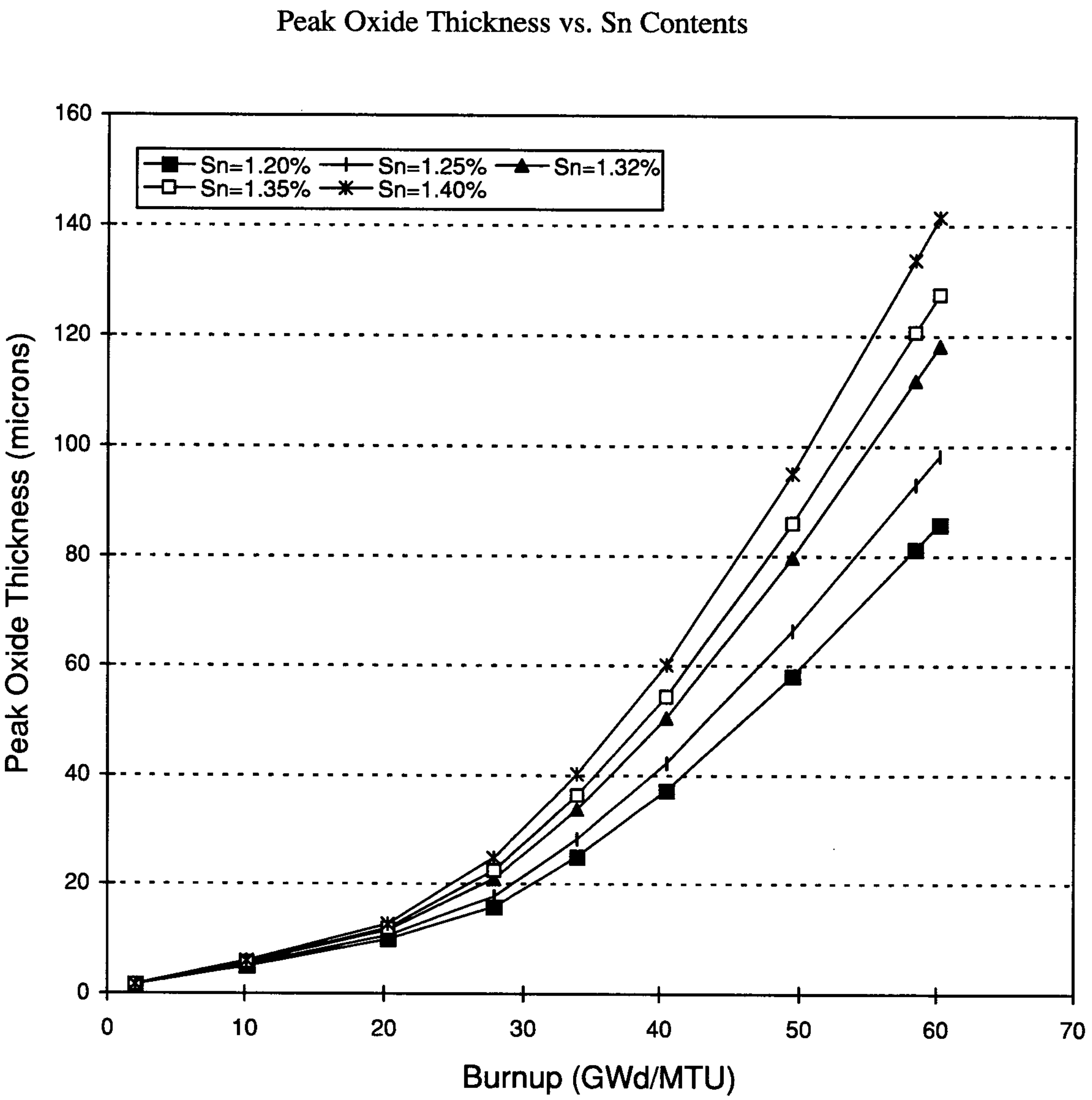


Figure 3-5
Peak oxide thickness vs. Sn contents for W fuel (Extended Chemistry Regime, Tin=291.4°C, Pavg=17.83 kW/m)

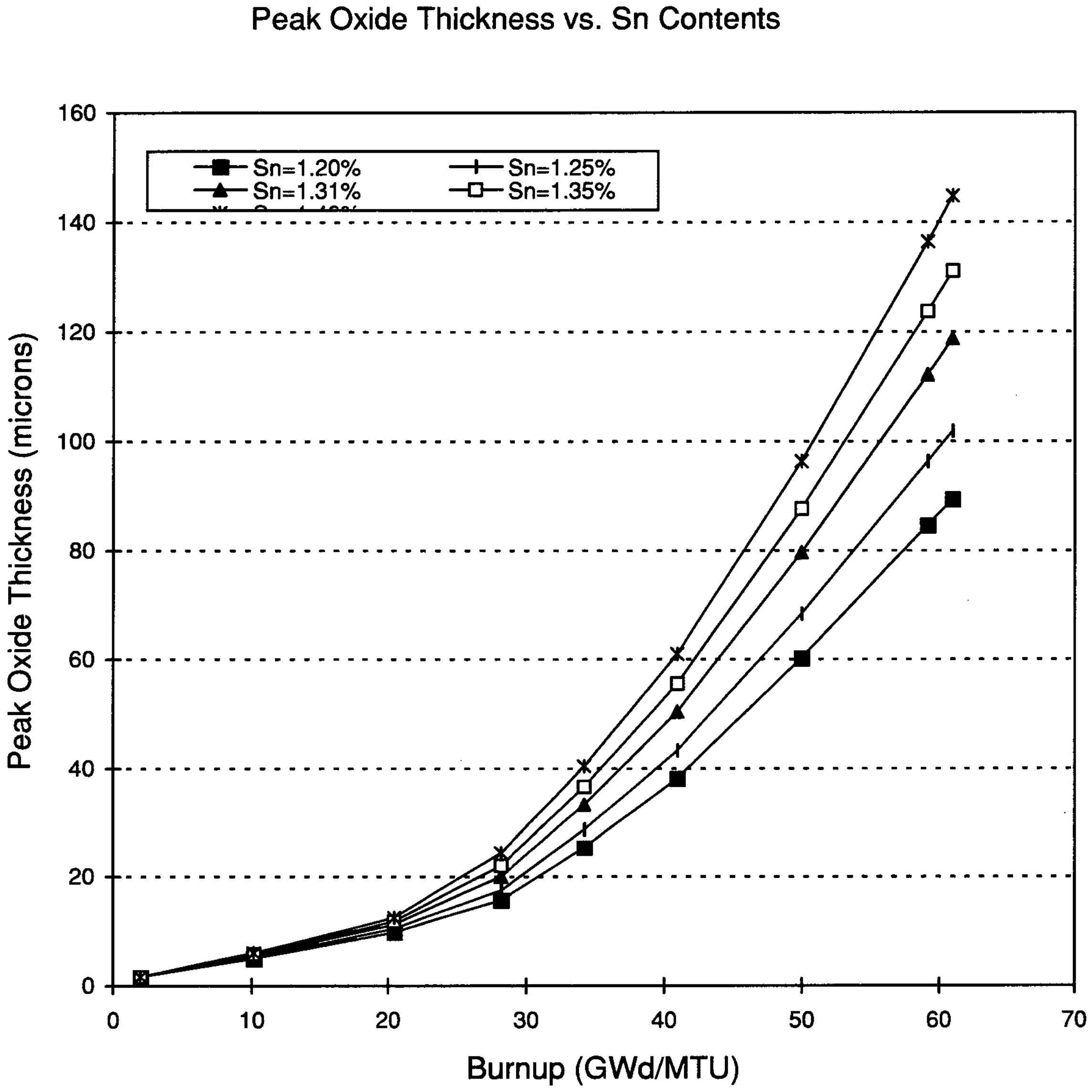


Figure 3-6
Peak oxide thickness vs. Sn contents for CE fuel (Extended Chemistry Regime, Tin=295.3°C, Pavg=17.69kW/m)

3.3 Power Peaking Factor and Local Boiling

The radial power peaking factor (RPF), an index of fuel thermal duty or heat flux, has a significant effect on the cladding corrosion rate by affecting the rod wall temperature. An increase in the Zircaloy cladding corrosion rate can occur under subcooled boiling conditions. The precise mechanism which causes the increase in the corrosion rate is not completely identified. However, high coolant lithium concentrations, combined with local boiling effect, have been known to cause accelerated Zircaloy corrosion [5,11]. When bubbles depart from the cladding surface, concentrated lithium is left behind in the liquid adjacent to the surface. Thus, to prevent an excessive Zircaloy corrosion rate, some limitations on the amount of local boiling and the lithium concentration in the coolant have to be placed. So far, there is insufficient data available to accurately determine the exact void fraction value which leads to accelerated corrosion.

Figures 3-7 and 3-8 show the peak oxide thickness for the increase in RPF. It can be seen that the cladding surface temperature exceeded the system saturation temperature of 344.83 °C at the upper region of the core if the RPF is greater than ~1.20 for Westinghouse fuel and ~1.25 for CE type fuel (Figures 3-9 to 3-12). Thus in the normal operating region, there happens significant local boiling in both plants since the peak cladding surface temperature exceeds 348 °C if RPF is greater than 1.35. The local boiling prevents rapid increase in fuel surface temperature induced by increase in RPF. Hence, the rate of increase in peak oxide thickness as a function of the RPF is lowered due to the onset of nucleate boiling (Figures 3-9 and 3-10). However, local subcooled boiling may result in the local concentration of boron and lithium which can be accelerated in the presence of crud deposits on the fuel cladding [5]. The result of the overall process may induce the Axial Offset Anomaly in the reactor core. Another potential effect of boiling is in concentrating lithium on the rod surface leading to accelerated corrosion. Without the presence of crud, boiling-induced corrosion acceleration has not been reported. Thus, the main issue related to boiling is crud deposition and its effect on fuel cladding corrosion via a chemical or thermal effect. Because of the sporadic nature of crud deposition the PFCC code has not yet incorporated the crud effect on cladding corrosion.

The overall accuracy of oxide thickness prediction using PFCC is within ±20 μm over the measured thickness range of 100 μm (Figure 3-13). As shown in Table 3-1, the sensitivity factors increased linearly as increase in RPF in both plants.

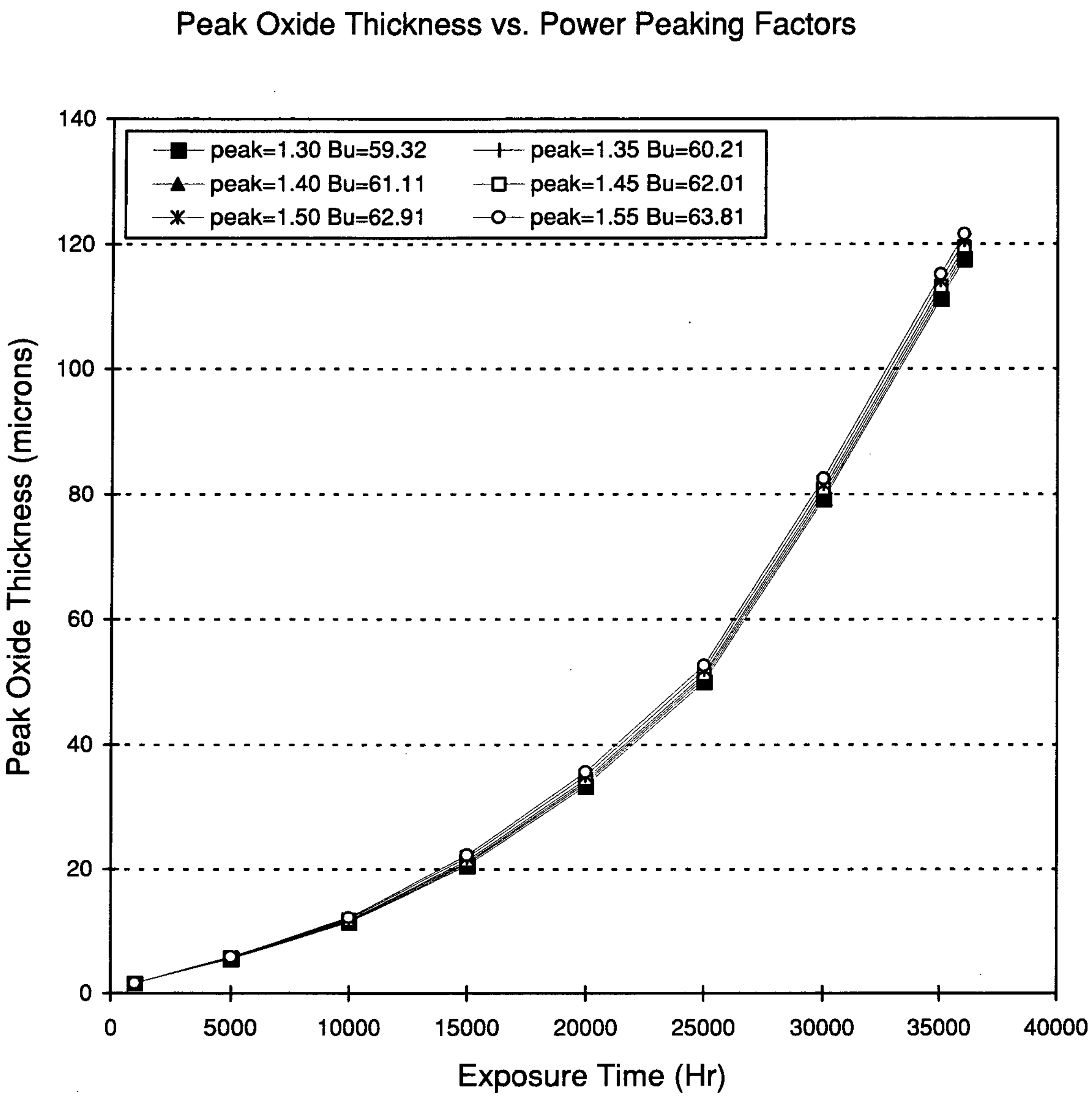


Figure 3-7
Peak oxide thickness vs. power peaking factors for W fuel (Extended Chemistry Regime, Tin=291.4°C, Pavg=17.83 kW/m)

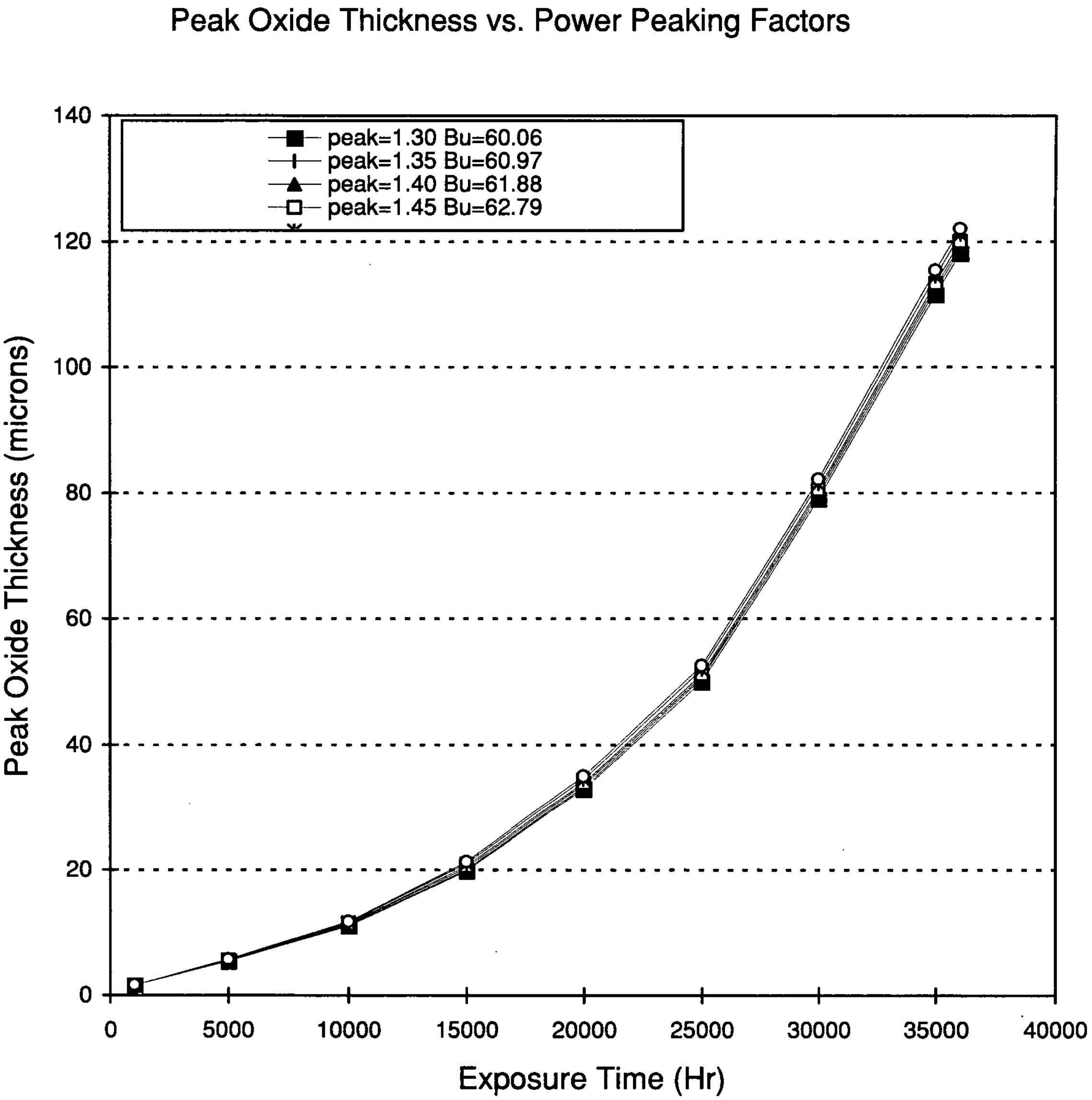


Figure 3-8
Peak oxide thickness vs. power peaking factors for CE fuel (Extended Chemistry Regime, Tin=295.3°C, Pavg=17.69kW/m)

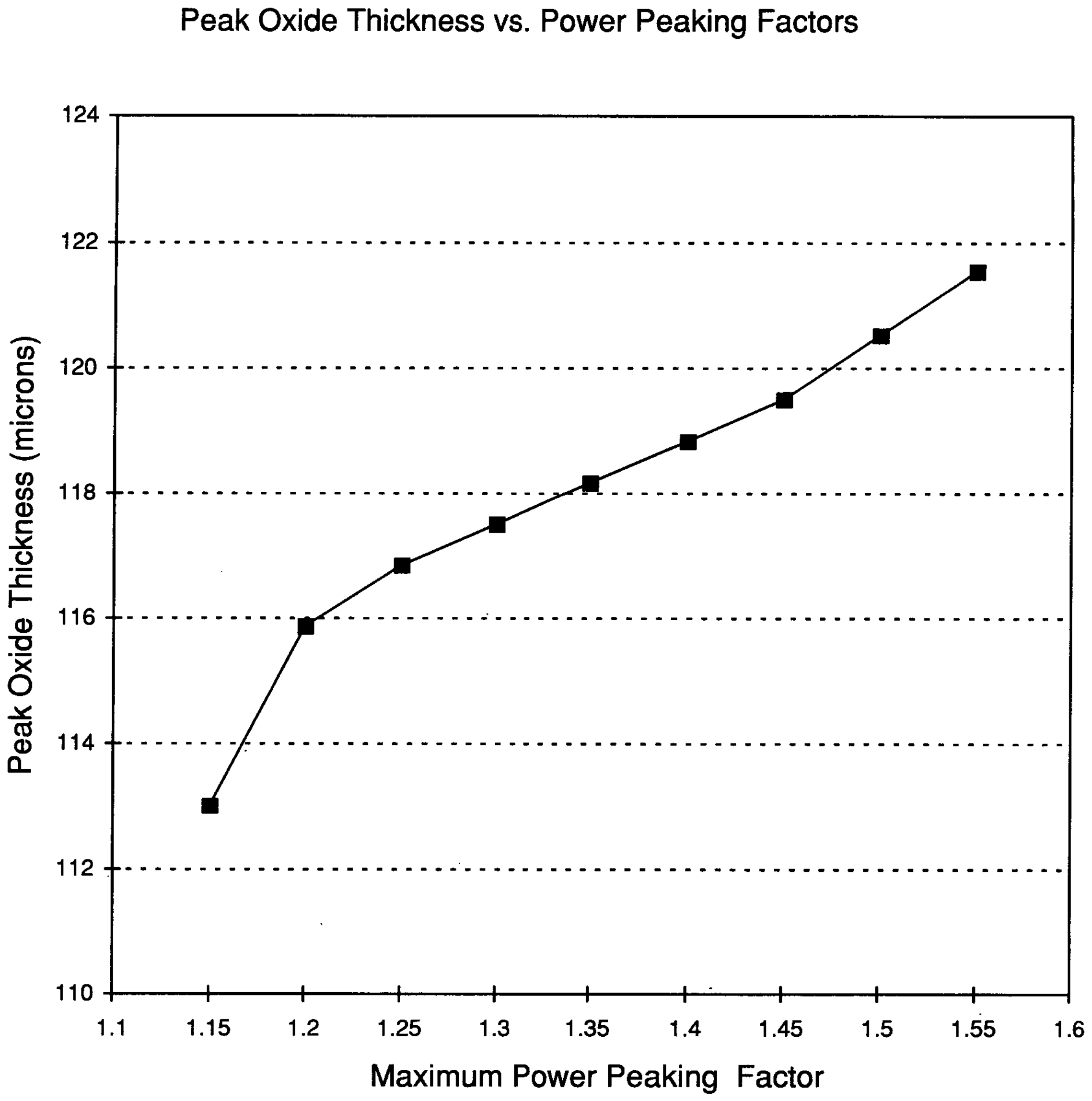


Figure 3-9
Peak oxide thickness vs. power peaking factors for W fuel (Extended Chemistry Regime, Tin=291.4°C, Pavg=17.83 kW/m)

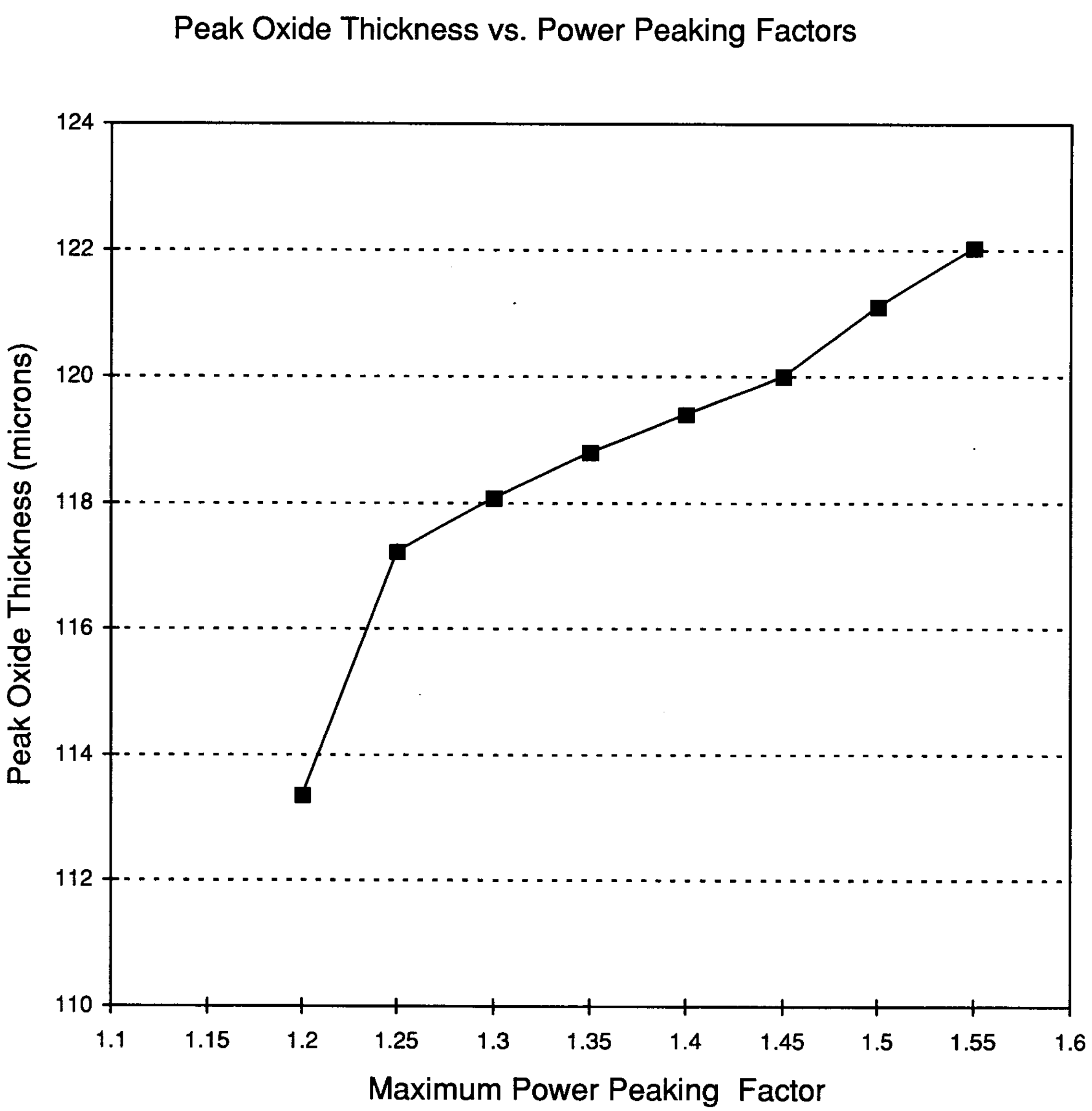


Figure 3-10
Peak oxide thickness vs. power peaking factors for CE fuel (Extended Chemistry Regime, Tin=295.3°C, Pavg=17.69kW/m)

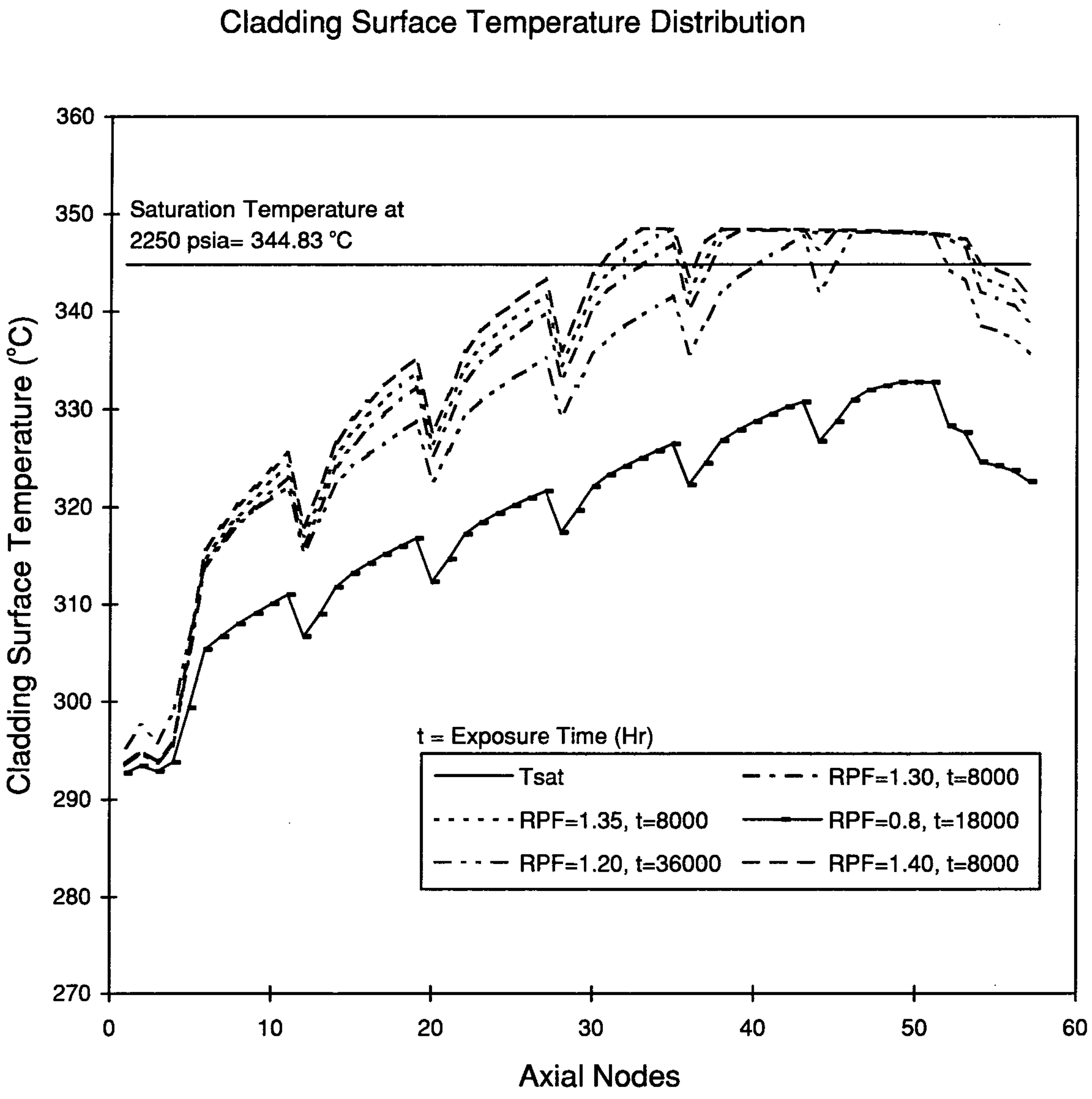


Figure 3-11
Cladding surface temperature distributions for W fuel (Extended Chemistry Regime, Tin=291.4°C, Pavg=17.83 kW/m)

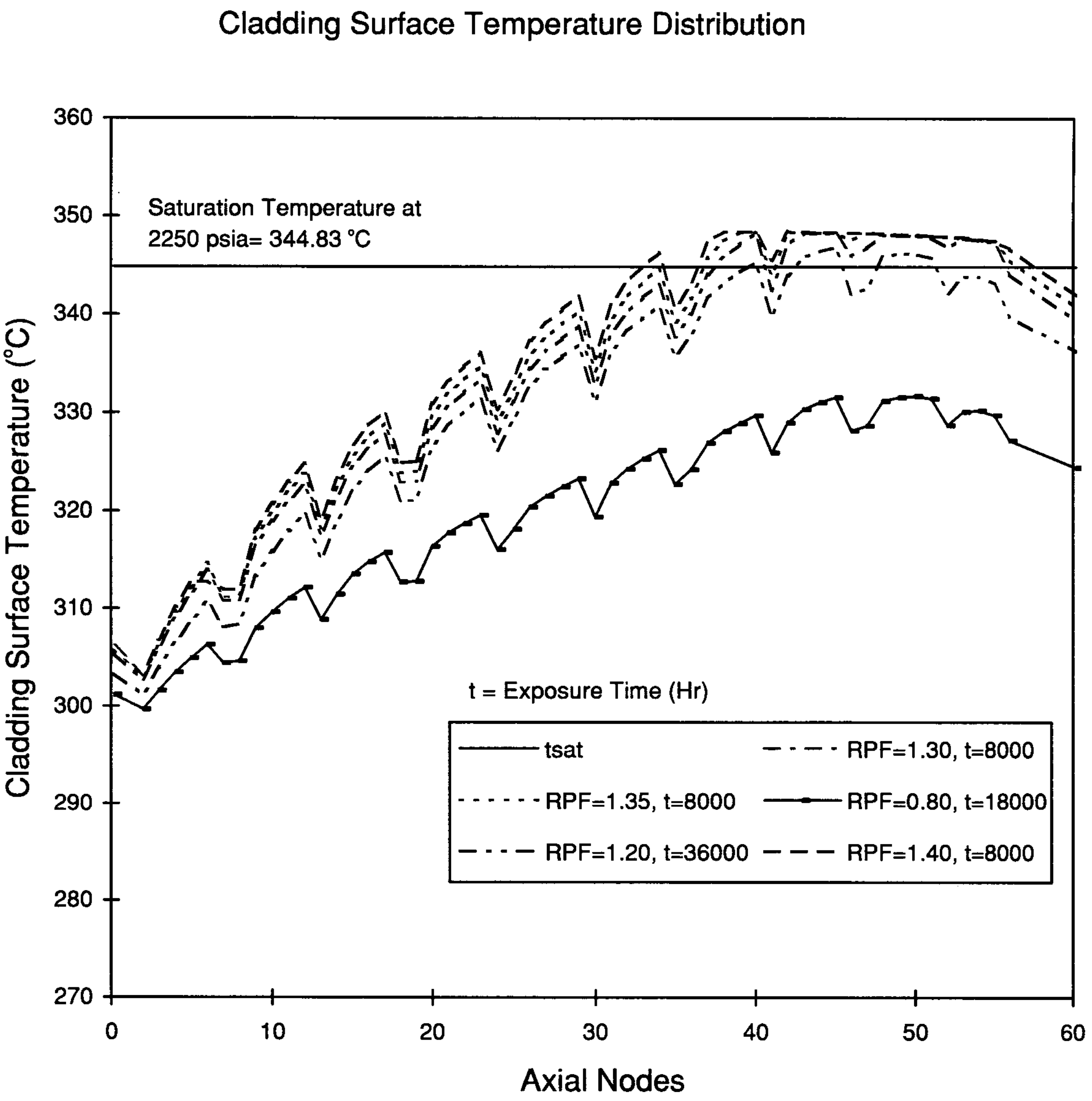


Figure 3-12
Cladding surface temperature distributions for CE fuel (Extended Chemistry Regime, Tin=295.3°C, Pavg=17.69kW/m)

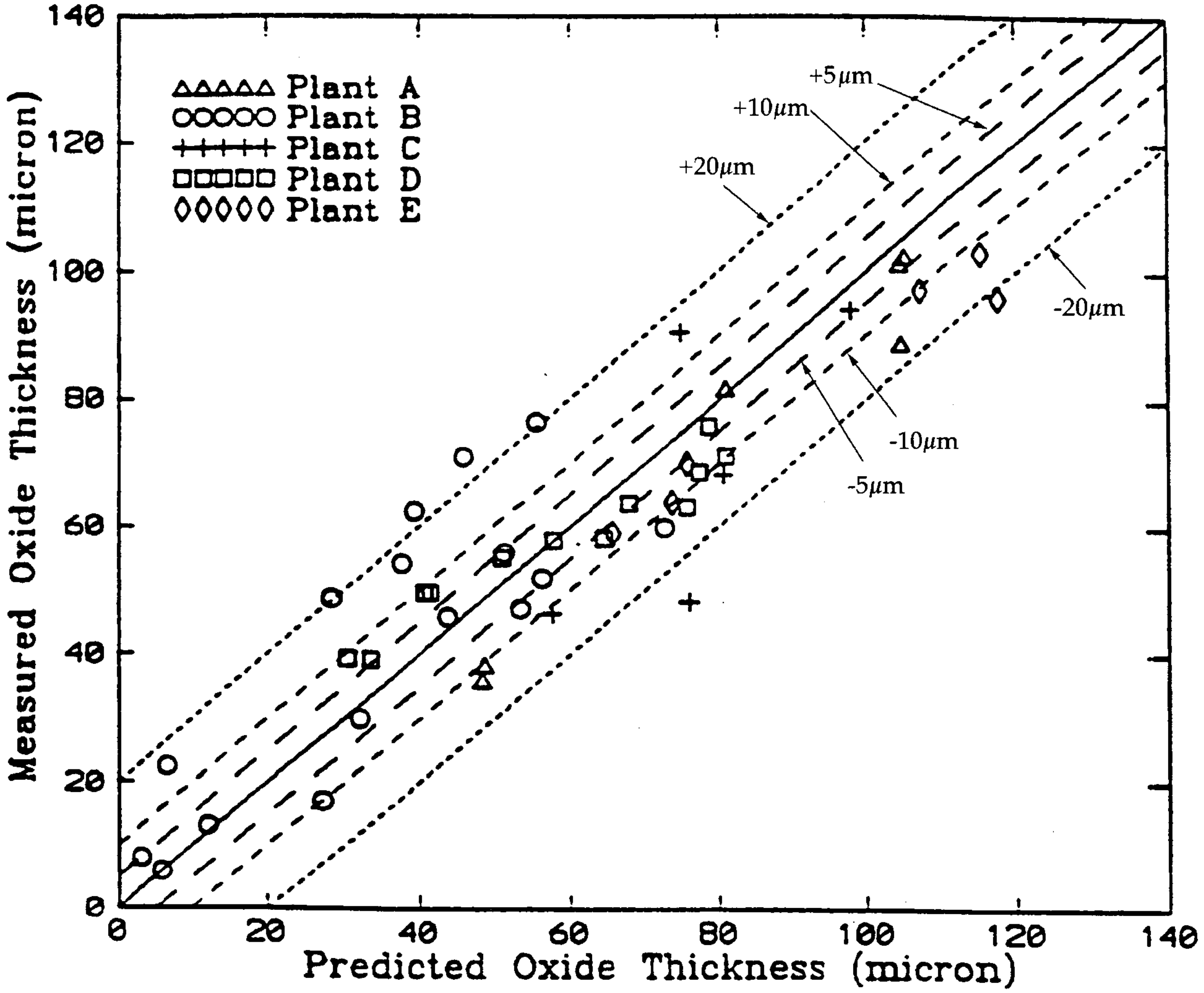


Figure 3-13
Measured vs. predicted peak oxide thickness of 45 benchmark and qualification rods [7]

3.4 Effect of Core Inlet Temperature and Flow Rate

The amount of nucleate boiling in a PWR depends on the plant specific system design, i.e., core power, coolant flow rate, inlet temperature, and core pressure. Plants A and B are operating under the same system pressure. However, plant B has greater values of the coolant flow rate, inlet temperature, cladding thickness and equivalent hydraulic diameter than those of plant A (Table 2-1 and 2-2). This gives somewhat different sensitivity to inlet temperature and mass flow rate for cladding corrosion in each plant. The higher inlet temperature and core mass flux can be compromised to give similar levels of sensitivity for cladding corrosion.

3.5 Effect of Axial Blanket

Westinghouse introduced the axial blanket to increase neutron economy into their Vantage-5 fuel. Axial blanket makes the power distribution be concentrated toward the center of the fuel assembly, which induces a rather increased axial power peaking. Figure 3-17 shows ~7% increase in the oxide thickness after introducing the axial blanket.

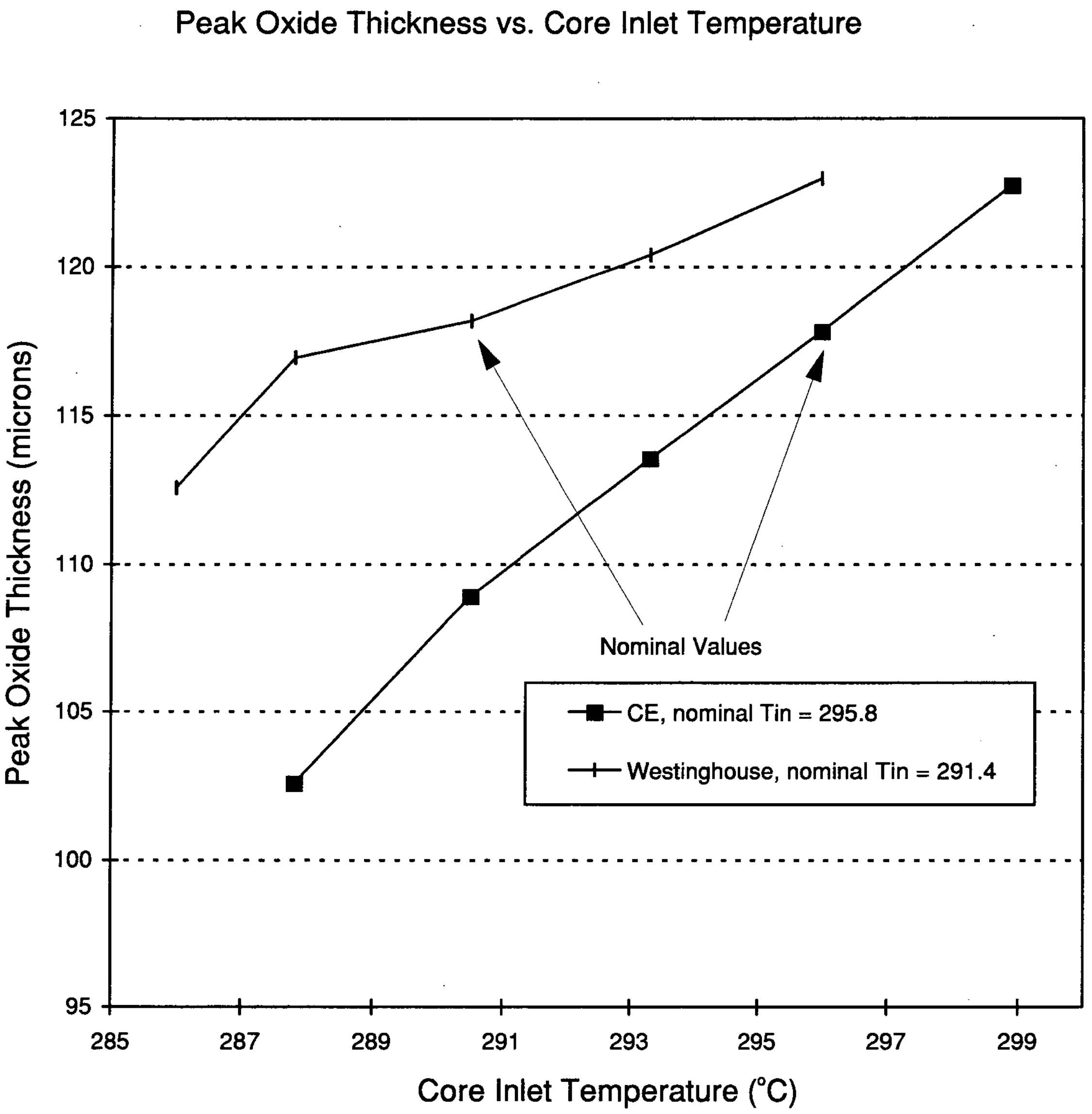


Figure 3-14
Peak oxide thickness vs. inlet temperature (Extended Chemistry Regime)

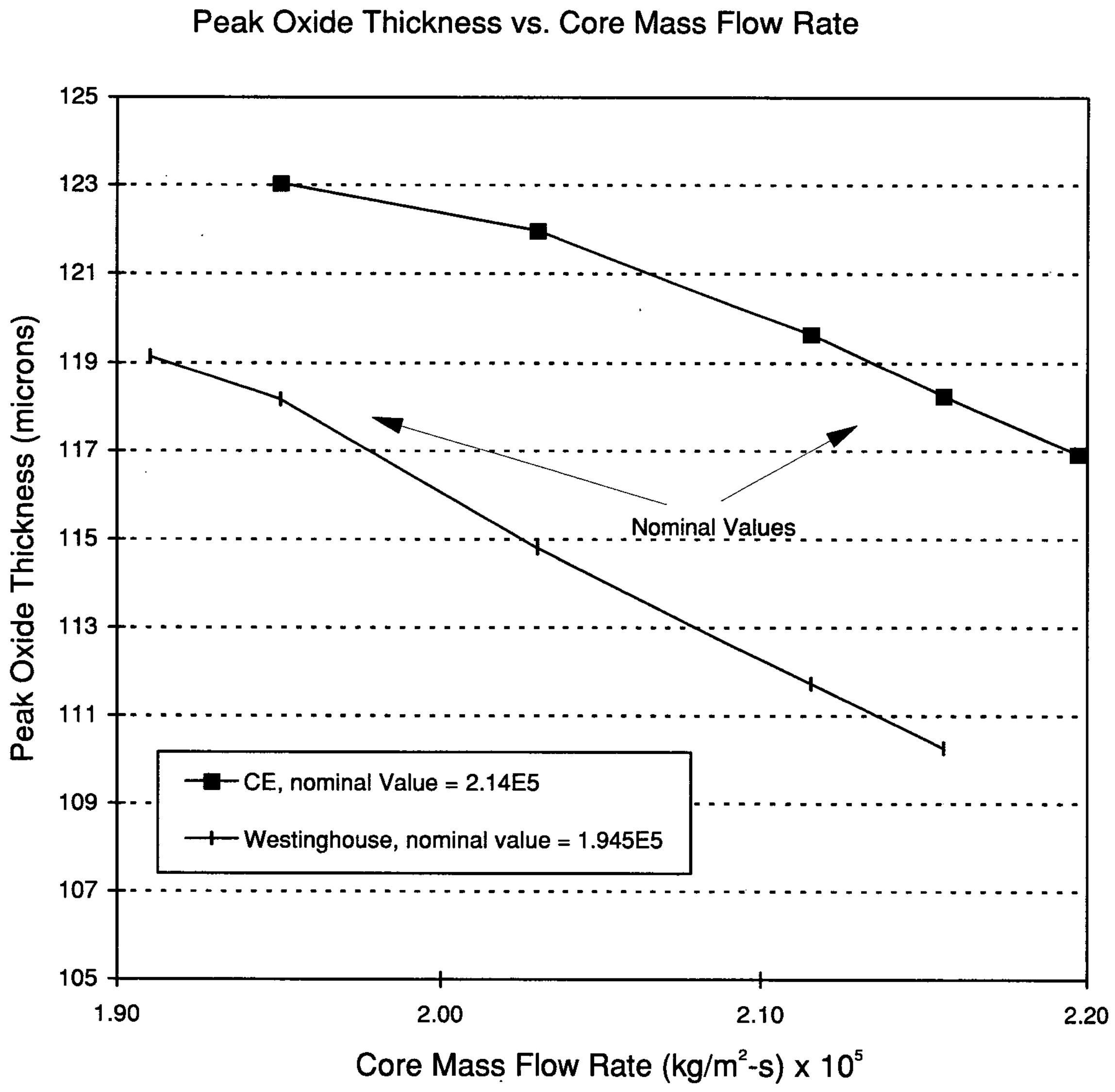


Figure 3-15
Peak oxide thickness vs. core mass flow rate (Extended Chemistry Regime)

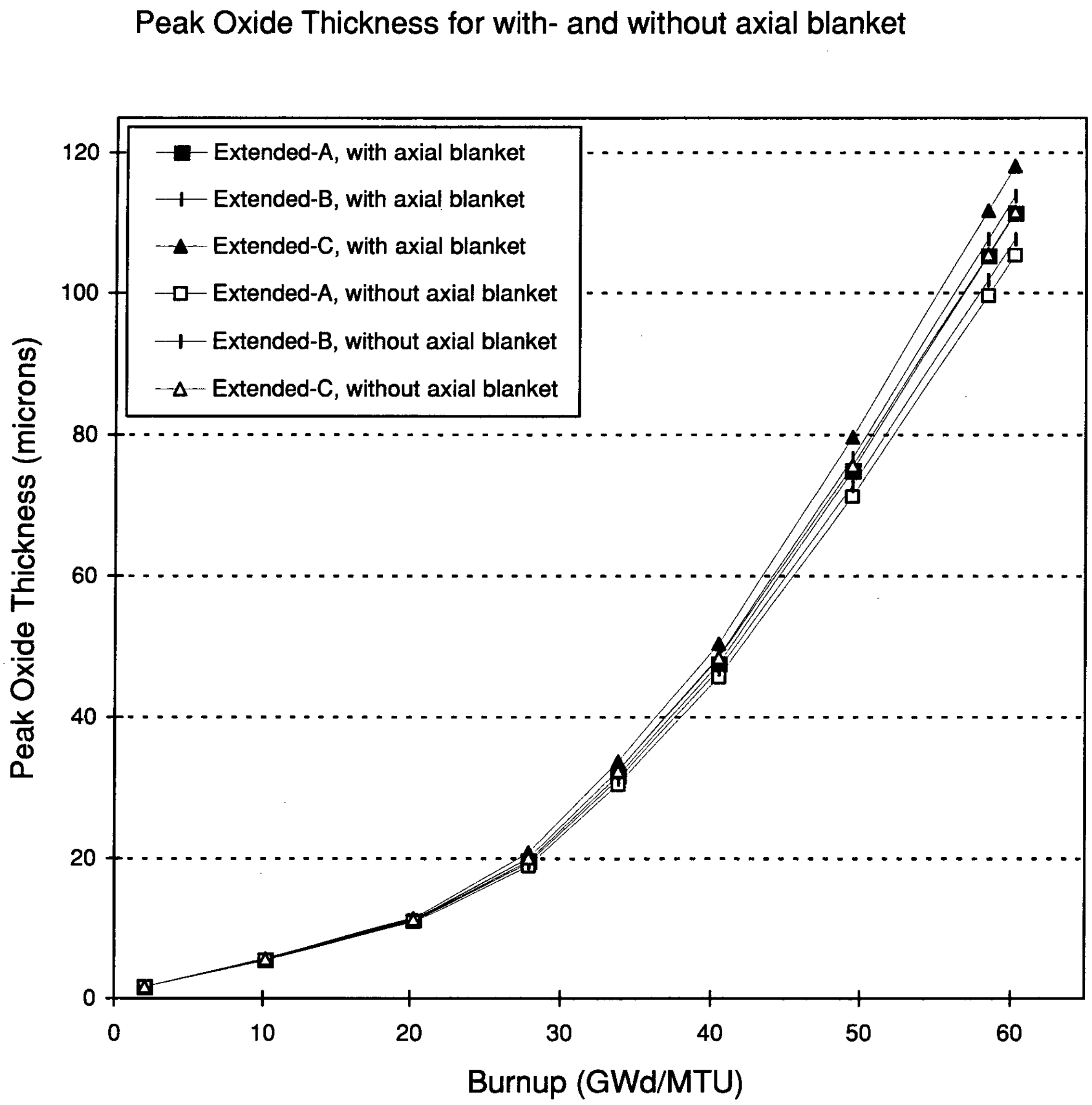


Figure 3-16
Peak oxide thickness vs. with- and without axial blanket for W fuel (Extended Chemistry Regime, Tin=291.4°C, Pavg=17.83 kW/m)

Table 3-1
Percent changes in oxide thickness for operating parameters

		Plant A (W)			**Plant B (CE)**		
Burnup (MWD/MTU)		25000	30000	36000	25000	30000	36000
Lithium chemistry		(%)					
elevated	-5%	-2.605	-2.676	-2.653	-2.529	-2.584	-2.542
	+5%	2.692	2.775	2.749	2.285	2.406x	2.424
extended	-5%	-0.852	-0.953	-0.787	-0.833	-0.905	-0.741
	+5%	0.852	0.966	0.795	0.813	0.905	0.741
Sn content		(%)					
	-5%	-15.481	-15.843	-16.019	-15.263	-15.209	-15.455
	+5%	16.848	16.834	17.339	16.947	16.805	17.357
RPF		(%)					
	-5%	-4.48	-5.457	-6.304	-4.361	-5.241	-6.086
	+5%	4.48	4.654	5.323	4.321	5.065	5.985
Core inlet temperature		(%)					
	-5 °F (-2.8 °C)	-3.449	-3.575	-3.58	-3.608	-3.695	-3.712
	+5 °F (+2.8 °C)	3.449	2.76	2.53	3.211	3.142	3.123
Core mass flux		(%)					
	-5%	2.636	2.095	1.98	2.438	2.577	2.685
	+5%	-2.498	-2.86	-3.123	-2.319	-2.69	-2.904

4

CONCLUSIONS AND SUMMARY

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1. The effects of reactor chemistry and operating variables on Zircaloy cladding corrosion are investigated for extended cycle operations of Westinghouse and CE plants using EPRI PFCC code.
2. The separate effects of the following variables which have been known to contribute to in-reactor corrosion of Zircaloy cladding under extended-cycle operation are discussed: lithium concentration, tin (Sn) content of cladding, power peaking factor and local boiling, core inlet temperature and mass flow rate.
3. A conservative power step time history was used. This bounding feature is introduced to consider the loading of the twice burned fuel into the core center. The axial power shape profile is assumed to have typical variation during a cycle operation of the reactor. Two types of lithium chemistry schemes were considered: Extended Cycle and Elevated Chemistry schemes. Initial lithium concentrations of 3~3.5 ppm for extended chemistry scheme and 2.5~3.5 ppm for elevated chemistry scheme were considered.
4. Predictions of oxide thickness have shown that there is not enough margin of cladding oxidation if the plant is operating under high inlet temperature and the rod burnup exceeds 55 GWD/MTU for extended chemistry and 50 GWD/MTU for elevated chemistry regime. For extended chemistry regime, the results indicate a small sensitivity factor of cladding corrosion on lithium concentration.
5. The axial temperature distributions of cladding surface show significant subcooled boiling in both type plants near core exit region since the peak cladding surface temperature exceeds the saturation temperature. Local subcooled boiling may result in locally concentrating lithium and boron. This process can be accelerated in the presence of crud deposits on the fuel cladding surface.
6. Based on the results of the prediction calculations, special concerns about cladding corrosion should be considered for extending the cycle length of PWRs beyond 18 months. For burnup exceeding 50-55 GWD/MTU, the low tin improved Zircaloy-4 cladding should be further improved.

5

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A

PFCC INPUT DATA

A.1 PFCC Input Values

The PFCC nominal values of input data for plant A and B are summarized in Table A1.

Table A-1
PFCC code inputs

	Plant A	Plant B
*UNI Data Block - Input Units Control		
Input units	Metric	Metric
Time units	Hours	Hours
*SYS Data Block - Reactor System Parameters		
System pressure	2.25E3 psia	2.25E3 psia
Inlet flow rate	1.945E5 kg/m^2-s	2.14E5 kg/m^2-s
Inlet temperature	291.4 °C	295.8 °C
*FUE Data Block - Fuel Rod Modeling Parameters		
Fuel pellet OD	0.8192 cm	0.8260 cm
Cladding OD	0.9500 cm	0.9700 cm
Cladding thickness	0.0572 cm	0.0640 cm
Active fuel length	365.8 cm	381.0 cm
Fuel density	95.00 %td	94.75 %td
Equiv. hyd. diameter	1.1773 cm	1.1973 cm
EFF heated diameter	1.1773 cm	1.1973 cm
*SPA Data Block - Spacer Grid Modeling Parameters		
Number of spacer grids	8	11
Spacer grid length	0 cm	0 cm
Local mix multiplier	1.000	1.000
Spacer grid axial locations (cm)	15.72, 77.77, 129.97, 182.17, 234.37, 286.56, 338.76, 391.06	10.95, 50.88, 90.81, 130.73, 170.66, 210.58, 250.51, 290.44, 330.36, 370.29, 410.22
*HEA Data Block - Heat Transfer Coefficient : All Default Values are Used		
*OXI Data Block - Oxidation Model Parameters		
Oxidation predictor	EPRI/SLI	EPRI/SLI
Oxide thermal conductivity	0.0150 W/cm-K	0.0150 W/cm-K

Table A-1 (cont.)
PFCC code inputs

	Plant A	Plant B
*HEA Data Block - Heat Transfer Coefficient : All Default Values are Used		
*OXI Data Block - Oxidation Model Parameters		
Oxidation predictor	EPRI/SLI	EPRI/SLI
Oxide thermal conductivity	0.0150 W/cm-K	0.0150 W/cm-K
C1	5.8876E+10	5.8876E+10
Q1	3.3663E+04	3.3663E+04
CLI	6.5000E-01	6.5000E-01
CFE	2.0000E-02	2.0000E-02
D3	2.0000E+00	2.0000E+00
C0	8.1980E+06	8.1980E+06
Q2	2.4825E+04	2.4825E+04
Q2U	9.1356E+03	9.1356E+03
CLI	6.5000E-01	6.5000E-01
CFE	2.0000E-02	2.0000E-02
CPHI	1.2000E-04	1.2000E-04
P0	2.4000E-01	2.4000E-01
SN (%)	1.32	1.31

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