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MILLER,C.L. Project Directorate I-2

SUBJECT: Forwards proprietary rept PL-NF-94-005-P, "Technical Basis for SPC 9X9-2 Extended Fuel Exposure at SSES" for review & approval.Rept provided in advance of forthcoming proposed OL amend request.Rept withheld (ref 10CFR2.790).

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MAY 31 1994

Director of Nuclear Reactor Regulation
Attention: Mr. C. L. Miller, Project Director
Project Directorate I-2
Division of Reactor Projects
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

SUSQUEHANNA STEAM ELECTRIC STATION
REQUEST FOR EXTENDED FUEL EXPOSURE LIMITS **Docket Nos. 50-387**
PLA-4147 FILES A17-2/R41-2 **and 50-388**

NOTE: THIS LETTER TRANSMITS PROPRIETARY INFORMATION PURSUANT TO 10CFR2.790, AND SHOULD BE WITHHELD FROM PUBLIC DISCLOSURE.

Dear Mr. Miller:

The purpose of this letter is to transmit the enclosed report entitled, "Technical Basis for SPC 9X9-2 Extended Fuel Exposure at Susquehanna SES" for NRC review and approval. This report is being provided in advance of a forthcoming proposed Operating License amendment which will request the report's inclusion as a reference in Technical Specification 6.9.3 as part of the basis for reload licensing of the Susquehanna units.

The report documents the basis for an increase in allowable exposure of SPC 9X9-2 fuel from 40 to 45 GWD/MTU, and was discussed with Mr. Shih-Liang Wu of NRR/SRXB on April 5, 1994. Our current plan is to submit the license amendment request later this year, after completion of confirmatory inspection activities on several lead use assemblies. The report is being submitted now in order to facilitate NRC review upon eventual receipt of the license amendment request referencing it. The first cycle which could be designed with the higher assembly exposures is Unit 1 Cycle 10; its design is currently scheduled to begin in March 1995. Therefore, PP&L requests NRC's decision on the proposed action prior to that time. We will keep you informed of any schedule changes as they occur.

Apo1
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Please note that Siemens Power Corporation (SPC) has identified information in the enclosed report that they consider to be proprietary. An SPC affidavit pursuant to 10CFR2.790 has been provided with the report. Accordingly, it is requested that the report be withheld from public disclosure.

This proposal represents a cost beneficial licensing action for PP&L in that it will decrease our fuel fabrication and spent fuel storage costs by approximately $1.25M per reload cycle. Questions should be directed to Mr. R.R. Sgarro at (610) 774-7914.

Very truly yours,

R. G. Byram

Attachment

cc: NRC Document Control Desk (original)
NRC Region I
Mr. G. S. Barber, NRC Sr. Resident Inspector - SSES
Mr. C. Poslusny, NRC Project Manager - OWFN
Mr. S. Wu, NRR/SRXB - OWFN

RRS:rrs\tah
4147.RRS

AFFIDAVIT

STATE OF WASHINGTON)
) ss.
COUNTY OF BENTON)

I, D. E. Hershberger being duly sworn, hereby say and depose:

1. I am Senior Engineer, Product Licensing, for Siemens Power Corporation, Nuclear Division, ("SPC"), and as such I am authorized to execute this Affidavit.

2. I am familiar with SPC's detailed document control system and policies which govern the protection and control of information.

3. I am familiar with the Pennsylvania Power & Light Company (PP&L) report entitled "Technical Basis for SPC 9X9-2 Extended Fuel Exposure at Susquehanna SES," referred to as "Document." Information contained in this Document has been classified by SPC as proprietary in accordance with the control system and policies established by SPC for the control and protection of information.

4. The Document contains information of a proprietary and confidential nature and is of the type customarily held in confidence by SPC and not made available to the public. Based on my experience, I am aware that other companies regard information of the kind contained in the Document as proprietary and confidential.

5. The Document has been made available to the U.S. Nuclear Regulatory Commission in confidence, with the request that the information contained in the Document will not be disclosed or divulged.

6. The Document contains information which is vital to a competitive advantage of SPC and would be helpful to competitors of SPC when competing with SPC.

7. The information contained in the Document is considered to be proprietary by SPC because it reveals certain distinguishing aspects of SPC product design and design methodology which secure competitive advantage to SPC for fuel design optimization and marketability, and includes information utilized by SPC in its business which affords SPC an opportunity to obtain a competitive advantage over its competitors who do not or may not know or use the information contained in the Document.

8. The disclosure of the proprietary information contained in the Document to a competitor would permit the competitor to reduce its expenditure of money and manpower and to improve its competitive position by giving it valuable insights into SPC product design and design methodology and would result in substantial harm to the competitive position of SPC.

9. The Document contains proprietary information which is held in confidence by SPC and is not available in public sources.

10. In accordance with SPC's policies governing the protection and control of information, proprietary information contained in the Document has been made available, on a limited basis, to others outside SPC only as required and under suitable agreement providing for nondisclosure and limited use of the information.

11. SPC policy requires that proprietary information be kept in a secured file or area and distributed on a need-to-know basis.

12. Information in this Document provides insight into SPC product design and design methodology developed by SPC. SPC has invested significant resources in developing the product and methodology as well as the strategy for this application. Assuming a competitor had available the same background data and incentives as SPC, the

competitor might, at a minimum, develop the information for the same expenditure of manpower and money as SPC.

THAT the statements made hereinabove are, to the best of my knowledge, information, and belief, truthful and complete.

FURTHER AFFIANT SAYETH NOT.

SUBSCRIBED before me this 13th

day of May, 1994.

Susan K. McCoy
NOTARY PUBLIC, STATE OF WASHINGTON
MY COMMISSION EXPIRES: 1/10/96

16

TECHNICAL BASIS FOR SPC 9X9-2 EXTENDED FUEL EXPOSURE

AT

SUSQUEHANNA SES

Prepared by: _____
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Reviewed by: _____
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Senior Scientist-Consulting

Approved by: _____
J. M. Kulick
Supervisor-Nuclear Fuels Engineering

May 16, 1994

PENNSYLVANIA POWER & LIGHT COMPANY

LEGAL NOTICE

This topical report was derived from information provided to Pennsylvania Power & Light (PP&L) by Siemens Power Corporation (SPC) and General Electric Company. This topical report is being submitted by PP&L to the Nuclear Regulator Commission (NRC) specifically in support of future proposed amendments to each of Susquehanna Steam Electric Station Unit 1 and 2 operating licenses to permit increasing the discharged exposure on SPC 9X9-2 fuel assemblies. The information in this report is true and correct to the best of PP&L's knowledge, information and belief.

Any use of this report or the information contained herein by anyone other than PP&L or the NRC is unauthorized. With regard to any unauthorized use, PP&L and it officers, directors, agents and employees make no warranty, either express or implied, as to the accuracy, completeness, or usefulness of this report or the information, and assume no liability with respect to its use.

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1.0 INTRODUCTION/SUMMARY

This report describes the technical basis for increasing the licensed discharge exposure limit on all Susquehanna Steam Electric Station (SSES) Siemens Power Corporation (SPC) 9X9-2 fuel assemblies from 40 GWD/MTU to 45 GWD/MTU. PP&L's desire to attain high assembly exposures is driven by fuel cycle economics and spent fuel storage considerations. The technical basis for the exposure extension includes on-site fuel inspections, fuel design analyses and evaluations, and an in-reactor fuel assembly extended exposure demonstration.

The in-reactor demonstration utilized four previously discharged SPC 9X9-2 fuel assemblies, with a bundle average exposure of 38 GWD/MTU. These assemblies underwent a detailed inspection prior to reinsertion, which included visual inspections and measurements of bundle growth, differential rod growth, rod diameter, oxide thickness, and channel engagement. The inspection results showed that the assemblies were in excellent condition and could be reinserted for a fourth cycle of irradiation to 48 GWD/MTU.

To support the in-reactor extended exposure demonstration, SPC performed additional mechanical design analyses to justify operation of the assemblies to an exposure of 48 GWD/MTU (Reference 1). Following approval by the NRC, the four assemblies were inserted into Unit 2 Cycle 6 (Reference 2) for their fourth cycle of irradiation. At the end of Cycle 6 (March 1994), the four demonstration assemblies successfully completed their extended duty to an exposure of 46,848 MWD/MTU. As part of a fuel sipping campaign during the Unit 2 sixth refueling outage, the four demonstration fuel assemblies were also vacuum sipped. As expected, the sipping confirmed that the four assemblies contained no failed rods.

In order to generically qualify the SPC 9X9-2 assembly for a maximum exposure of 45 GWD/MTU, SPC performed additional mechanical design analyses. These analyses assumed a slightly increased LHGR limit for additional operating flexibility at power uprate conditions. The revised 9X9-2 LHGR limits used in these analyses are the same as the limits for other SPC fuel designs (e.g., 9X9-5 and 9X9-9X). The revised 9X9-2 LHGR limits as a function of exposure include the steady state mechanical design LHGR limit (Figure 1) and Protection Against Power Transients limit (Figure 2). The results of the mechanical design analyses show that all the design criteria (References 1 and 4) are met for assembly exposures to 45 GWD/MTU.

In total, all of the fuel inspections, analyses, and the in-reactor demonstration show that the current SPC 9X9-2 fuel assemblies are capable of operating safely to 45 GWD/MTU at power uprate with increased core flow conditions at SSES. As an additional confirmation, PP&L plans to perform a detailed re-inspection of the extended exposure demonstration assemblies following the Unit 2 sixth refueling outage.

2.0 IN-REACTOR FUEL ASSEMBLY PERFORMANCE

As described in Section 1.0, four previously irradiated SPC 9X9-2 fuel assemblies were selected for the in-reactor demonstration of extended exposure operation. These fuel assemblies were the four highest exposed SPC 9X9-2 assemblies at SSES. The assemblies were selected in order to achieve an end-of-life exposure greater than 45 GWD/MTU during Unit 2 Cycle 6. These fuel assemblies, which were initially inserted into Unit 2 Cycle 2, were discharged at the end of Cycle 4 after three cycles of operation. In each cycle of operation, the assemblies were located in symmetric core locations and each accumulated an assembly average exposure of approximately 38,030 MWD/MTU over the three cycles of operation.

The following sections contain a summary of results from the first fuel inspection at an exposure of 38,030 MWD/MTU as well as a description of the criteria which will be applied to the results of the inspection of the four demonstration fuel assemblies following their extended exposure operation to 46,848 MWD/MTU.

2.1 Fuel Inspection Results at 38,030 MWD/MTU

2.1.1 Background

Prior to the reinsertion of the four SPC 9X9-2 fuel assemblies for extended irradiation, detailed inspections were performed to determine the performance of the fuel assemblies near the current licensing limit (40 GWD/MTU) and to obtain the characterization data necessary to support higher fuel assembly exposures. These inspections included a channel-to-lower tie plate seal spring gap measurement on two assemblies (X21-037 and X21-287) and a detailed rod and bundle inspection on the two other assemblies (X21-038 and X21-288). Only two assemblies were disassembled for the detailed rod inspection in order to limit the number of irradiated fuel rods being handled. The other two (undisturbed) assemblies will provide a frame of reference for the post-extended irradiation inspection. The detailed rod and bundle inspection included visual inspections and measurements of the fuel rod oxide thickness, fuel rod profilometry, fuel rod differential growth, and bundle growth, as discussed in more detail below.

For the detailed rod inspections, twenty fuel rods and two water rods were selected from fuel assemblies X21-038 and X21-288. The twenty fuel rods were composed of six UO_2 rods (i.e., standard fuel) and four NAF rods (i.e., Neutron Absorbing Fuel: fuel rods containing gadolinia) from each of the two assemblies.

2.1.2 Visual Inspection

Tie Plate Inspection

The upper and lower tie plates were inspected on the assemblies. No unusual marks, wear, or damage were observed. The rod-to-upper tie plate engagement appeared adequate for further irradiation which was confirmed by subsequent measurements. The lower tie plate seal appeared to be normal and all the finger springs were in place.

Spacer Inspection

The visual inspection of the fuel spacers showed that they all appeared to be in good condition, with no excessive oxide coverage, wear, or damage. Based on the normal marks on the spacer 'tubs', the spacers were probably not in hard contact with the channel. Also, there was no problem in dechanneling or channeling the bundle.

Fuel Rod Inspection

A visual inspection of the fuel rods, while the rods were in the bundle, showed that the fuel rods had a substantial layer of soft red crud (mainly iron oxide) on the surface. The higher power regions of the fuel rods appeared to have more crud, which is as expected.

The visual inspection of the individual fuel rods, which were removed from the bundle, involved first cleaning the rod at the high pressure water wash station to remove the loose crud. The fuel rods appeared to have a thin layer of oxide on the surface of the rod varying from black on the first and eighth spans (low rod power regions) to white on the intermediate spans (high rod power regions). Very little nodular corrosion was observed on any of the rods examined. The upper and lower end cap welds appeared normal with no unusual wear or corrosion. The regions of spacer contact on the rods showed some superficial markings which were not any larger than expected. Overall, the fuel rods were in excellent condition.

2.1.3 Channel Gap Measurement

The distance between the bottom edge of the channel and the bottom of the lower tie plate seal spring (referred to as the "channel gap") was measured on two assemblies to determine the remaining channel-to-lower tie plate engagement. The channel engagement is defined as the vertical distance between the bottom of the channel and the point on the seal spring where positive engagement is lost. However, for the purpose of these fuel inspections, channel engagement was conservatively defined as the distance to the top of the land area on the seal spring (disengagement will not occur until some distance above the land area).

The measurements showed that the channel gap was 1.28 inches and 1.24 inches for assemblies X21-037 and X21-287, respectively. Based on nominal as-fabricated dimensions, the remaining channel engagements (to top of land area) were greater than or equal to 0.52 inch for assemblies X21-037 and X21-287. Based on a maximum bundle growth of 0.29 inch and conservatively assuming no channel growth, there was clearly sufficient channel engagement for another cycle of irradiation.

2.1.4 Rod Oxide Measurement

Rod oxide measurements were taken on a total of twenty fuel rods and two water rods after the loose crud was removed. The axial oxide profile was measured on twelve UO_2 rods, eight NAF rods, and two water rods from the two assemblies.

The measured maximum peak oxide thickness was 25 microns on the UO_2 rods, 21 microns on the NAF rods, and 32 microns on the water rods. The average of the peak oxide measurement in each of the eight spans was 11.2 microns on the UO_2 rods, 10.2 microns on the NAF rods, and 19.3 microns on the water rods. Based on the oxide measurements, the SSES fuel rods have a relatively thin oxide layer for such high assembly exposures in comparison to the average of the SPC data base.

2.1.5 Rod Profilometry Measurement

Individual rod diameter profilometry measurements were taken on a total of twenty fuel rods (twelve UO_2 rods, eight NAF rods) and one water rod to gather fuel rod creepdown and ovality data. Correction of the as-measured diameters for the oxide thickness was performed to determine the true fuel rod profile.

The results of the rod diameter measurements show that no ridging occurred on any of the fuel rods. Therefore, the fuel pellets were not in hard contact with the cladding for any extended length of time.

The fuel rod ovality was determined by taking the difference between the maximum diameter and the minimum diameter in each span. The results show that all the fuel rods have low ovality. The maximum ovality was 1.9 mils and the average ovality was 1.3 mils.

The fuel rod diametral creepdown was calculated to determine how much the fuel rod cladding crept-down (i.e., shrunk) onto the fuel pellets. The measurements showed that, after three cycles of irradiation, the fuel rod diametral creepdown over the high power spans averaged 0.56% for the UO_2 rods and 0.52% for the NAF rods. This amount of cladding creepdown is slightly below the average of all SPC 9X9 data.

2.1.6 Fuel Rod and Bundle Growth Measurements

Fuel rod and assembly growth measurements were performed on two assemblies (X21-038 and X21-288). The results of the individual rod measurements show that fuel rods grew an average of 0.95 inch, Spacer Capture Rods (SCRs) grew an average of 0.79 inch, and Water Rods (WRs) grew an average of 0.85 inch. The results of the bundle measurements show that bundles X21-038 and X21-288 grew 1.025 inch and 0.989 inch, respectively which is consistent with the SPC data base.

From the rod and assembly measurements, the fuel rod to upper tie plate engagement was determined. The fuel rod engagement is defined as how far the fuel rod is inserted into the upper tie plate less the lengths of the tapered portion of the upper end plug and the inlet chamfer on the bottom of the upper tie plate. The results show that the minimum engagement was 0.653 inch for the fuel rods, 0.611 inch for the SCRs, and 0.651 inch for the WRs which is consistent with the SPC data base.

2.2 Fuel Inspection Criteria For Assemblies at 46,848 MWD/MTU

This section contains the criteria that will be used to confirm the performance of the four extended exposure demonstration fuel assemblies. The four demonstration assemblies have completed a successful fourth cycle of irradiation at the end of Unit 2, Cycle 6, and reached an exposure of 46,848 MWD/MTU. The following are the five criteria (provided by SPC) that will be used for the inspection to confirm that SPC 9X9-2 fuel assemblies have the capability to safely attain an exposure of 45,000 MWD/MTU.

1. The maximum rod oxide thickness is less than 3 mils.

2. The fuel rod is not disengaged from the upper tie plate (i.e., the tapered portion of the upper end plug is above the inlet chamfer on the bottom of the upper tie plate).

3. The fuel rod diameter and ovality are consistent with the SPC data base.

4. The fuel channel is in positive contact with the lower tie plate seal.

5. The rod-to-rod spacing shows no unusual gap closure based on a visual inspection.

3.0 GENERIC FUEL MECHANICAL DESIGN ANALYSIS

A mechanical design analysis was performed by SPC to support extension of the SPC 9X9-2 licensed assembly average exposure from 40 GWD/MTU to 45 GWD/MTU (References 1 & 5). The new mechanical design analysis includes a revised 9X9-2 LHGR limit versus exposure curve (Figure 1), which is the same as the SPC 9X9-5 LHGR limit versus exposure curve, but extended to cover an assembly exposure of 45 GWD/MTU. In addition, the extended exposure mechanical design analysis includes the effects of power uprate and increased core flow conditions.

The mechanical design analysis provides evaluations of the following fuel rod cladding parameters with respect to their design criteria: differential expansion and growth, creep collapse, steady state strain, corrosion and hydrogen absorption, steady state stress, temperature, transient strain, and cyclic fatigue. The fuel rod spacing and bow, pellet steady state temperature, fuel rod internal pressure, water rod design, grid spacer design, compression spring design, and the lower tie plate seal were also evaluated.

The results of the analysis support irradiation of the SPC 9X9-2 fuel to the following exposures:

	MWD/MTU
Fuel Assembly	45,000
Fuel Rod	51,600
Peak Pellet	60,000
Planar Exposure	57,300

3.1 Design Criteria

The 9X9-2 fuel assemblies are designed to satisfy the objectives described in the SPC generic mechanical design report (Reference 4) and the NRC Standard Review Plan Section 4.2. The fuel design criteria are based on material design limits and address fuel damage mechanisms with appropriate limits. For each criterion, the mechanical design limit and the results of the analyses are stated in each subsection below. The following nominal reactor conditions were used in the mechanical design analysis.

Nominal Reactor Conditions	Value
Core Power	3441 MWt
Core Coolant Operating Pressure	1065 psia
Rated Core Flow	108×10^6 lbm/hr

3.2 Differential Thermal Expansion and Growth Allowance

The maximum predicted fuel assembly growth is 1.6 inches. This growth will not cause operational limitations or interfere with the surrounding reactor internals.

The criterion on the fuel rod thermal expansion and growth is that the fuel rods do not disengage from the upper tie plate. The maximum predicted differential growth between the bundle (i.e., tie rods) and the standard fuel rods at 45 GWD/MTU is 0.29 inch. The corresponding predicted differential growth between the bundle and the water rods is 0.44 inch. The nominal engagement of the standard fuel rods and water rods into the upper tie plate is 0.853 inch and 0.864 inch, respectively. The corresponding root mean squared maximum manufacturing tolerances are 0.132 inch and 0.086 inch, respectively. Therefore, after subtracting the differential growths and manufacturing tolerances from the BOL engagements, the predicted EOL engagements (0.43 inch and 0.34 inch for the standard fuel rods and water rods, respectively) are sufficient to maintain engagement to 45 GWD/MTU.

3.3 Cladding Creep Collapse

A design criterion on the cladding is that no significant axial gaps will form in the fuel column during fuel pellet densification. Since the majority of the fuel densification occurs early in life, the design basis requires that the fuel column, under the force of the plenum spring, will remain compressed at initial exposures. A second criterion is that the reduction in the pellet-to-cladding gap due to cladding circumferential creep and ovality shall not exceed the initial cold gap up to a rod average exposure of 6,000 MWD/MTU.

An evaluation of the pellet column shows that no axial gaps will form and, therefore, the pellet column will support the cladding. Thus, the cladding will be prevented from collapsing during pellet densification.

An analysis was performed with the revised 9X9-2 LHGR limits to demonstrate that the reduction in the radial gap does not exceed the initial cold radial gap. The results show a cold radial gap of 0.00275 inch (which includes conservative allowance for manufacturing tolerances) at 0.0 MWD/MTU and 0.0008 inch at 7,254 MWD/MTU. Therefore, the reduction in the cold radial gap does not exceed the initial cold radial gap to a rod exposure of 6,000 MWD/MTU.

3.4 Fuel Rod Spacing and Bow

The fuel rod spacing and bow shall be accommodated throughout the design life of the fuel bundle to support the reactor operating thermal limits (i.e., no reduction in MCPR). Based on rod-to-rod spacing data, the calculated minimum spacing at 45 GWD/MTU is 0.094 inch which is greater than the design criterion of 0.09 inch.

3.5 Cladding Steady-State Strain

The design steady-state cladding strain criterion is that the circumferential plastic strain shall not exceed 1.0% at end of design life. The analysis shows that the calculated EOL strain is 0.35% which is well below the limit of 1.0%.

3.6 Cladding Corrosion and Hydrogen Absorption

The fuel design basis for cladding corrosion and crud buildup is to prevent (1) significant degradation of cladding strength and (2) unacceptable temperature increases. Because of the thermal resistance of corrosion and crud layers, formation of these products on the cladding results in elevation of the temperature within the fuel as well as the cladding. SPC uses a cladding outer surface temperature limit for corrosion that is specified in Section 3.10.

The metal loss due to corrosion at EOL is accounted for in the cladding stress calculation. A maximum of 78 microns of oxidation (equivalent 2 mils wall thinning) was conservatively assumed in the fuel rod stress calculation. The maximum expected EOL oxide corrosion thickness (based on SPC's conservative correlation) is less than the design criteria of 78 microns. Based on the inspections, the maximum measured oxide was 25 microns at 38 GWD/MTU, which is less than that predicted by SPC's conservative correlation. Consequently, corrosion of the fuel rod cladding will not be limiting at 45 GWD/MTU.

A calculation of the wall thinning of the water rod and SCR was also performed using the same corrosion model as applied to the fuel rods. The calculation shows that wall thinning is less than the 2 mil criterion used in the mechanical design analysis.

The hydrogen absorption criterion is that the cladding hydrogen content shall not exceed 300 ppm, on a cladding weight basis, under the most adverse projected power conditions within chemistry limits. The calculated 95/95 upper limit EOL cladding hydrogen concentration is less than the design criterion.

3.7 Steady-State Stress

The steady-state stress analysis is performed at BOL and EOL for both hot and cold conditions. Each individual stress is calculated inside and outside the cladding at both mid-span and spacer level. The applicable stresses at each level are then combined to get the maximum stress intensities. The results of the analysis show that the maximum ratio of the stress intensity to the design limit for any cladding member is less than the limit of 1.0.

A stress analysis was also performed for the fuel rod end cap as the maximum temperature gradient occurs at this location. The mechanical stress is caused by (1) the pressure differential across the cladding and (2) the axial loads due to the pellet stack weight and the plenum spring force. The thermal stress is caused by the temperature gradient between the external surface of the end cap and the heat from the pellets. The analysis shows that the maximum ratio of the stress intensity to the design limit of any lower end cap member is less than the limit of 1.0.

3.8 Cladding Transient Strain

The cladding transient circumferential strain criterion is that the strain shall not exceed 1.0% for all power levels below the curve of LHGR versus Exposure defined in Figure 2, "Revised 9X9-2 Limits for Protection Against Power Transients". In the past, these limits have been referred to as Protection Against Fuel Failures limits. The cladding circumferential strain is defined as the uniform strain which includes both elastic and inelastic strains. The results show that the transient cladding strain does not exceed 1.0% for LHGRs at or below the Figure 2 LHGR curve.

3.9 Pellet Steady State Temperature

The fuel pellet centerline temperature shall not exceed the melting temperature for all power levels. The fuel pellet centerline temperature is calculated at an overpower condition as a check against the occurrence of centerline melt in an anticipated operational transient event. The calculation shows that the peak centerline temperature at a 135.1% overpower is less than 4237°F which is well below the UO_2 melting point of 5039°F. Therefore, the fuel pellet centerline temperature will not exceed the UO_2 melting temperature.

3.10 Cladding Steady-State Temperatures

The steady-state cladding temperature is determined to provide assurance that the corrosion performance of the fuel rod is acceptable. For this calculation, the power history that is used in other calculations (such as internal pressure) results in maximum cladding temperatures below the limits (see values below).

Parameters	Value	Limit
• Cladding Inside Diameter Temperature, (°F)	<696	850
• Volumetric Average Temperature, (°F)	<635	750
• Cladding Outside Diameter Temperature, (°F)	<576	625

3.11 Cladding Cyclic Fatigue

The criterion on the cladding cyclic fatigue is that the cumulative usage factor for cyclic stress for all important loading conditions shall not exceed 0.67. The loading (i.e., duty) cycles encompass the normal reactor operations expected over the design life of the fuel. The new duty cycles, to account for the extended exposure, were calculated by increasing the current duty cycles by a factor of 1.125 (45/40). The total usage factor was calculated to be 0.06 which is well below the limit of 0.67 for cyclic failure.

3.12 Fuel Rod Internal Pressure

The fuel rod end-of-life internal pressure is required to be less than 800 psi above the operating core pressure (1065 psia). The calculation shows the rod internal pressure will be less than 1865 psia. The second criterion is that the pellet to cladding gap remains closed (once closed). The calculated pellet to cladding gap, onced closed, remains closed out to 51.6 GWD/MTU (maximum rod exposure at a bundle exposure of 45 GWD/MTU).

3.13 Water Rod Design

In a 9X9-2 fuel bundle, there are two water rods, one of which is a Spacer Capture Rod (SCR). The water rod criterion is that the geometry must be maintained to assure proper coolant flow throughout the fuel assembly.

The stress on the water rod from restrained bow (between the spacers) and flow induced vibration was calculated in the same manner used for a fuel rod. Because the grid cell characteristics are identical for the water rod and fuel rod, the axial friction force for the two rods was assumed to be the same. The calculation shows that the stresses on a water rod are bounded by the stresses on a fuel rod, and therefore, proper geometry will be maintained.

3.14 Grid Spacer Design

The function of the grid spacer assembly is to (1) maintain separation of the fuel rods and water rods, (2) provide mechanical strength to the fuel assembly, and (3) restrain fuel rod bowing and vibration throughout the design life of the fuel assembly. The minimum EOL spring force, which occurs due to irradiation induced stress relaxation, is designed to accommodate flow-induced vibration.

An analysis was performed to demonstrate that the minimum EOL spring force is greater than the calculated required force of 0.23 lbf. Based on an initial BOL minimum spring force of 4 lbf, the minimum EOL spring force is calculated to be 0.40 lbf which is higher than the limit. Therefore, there is sufficient spring force to satisfy the spacer assembly criterion.

3.15 Compression Spring

The criterion on the compression spring is that the spring force must be sufficient to support the upper tie plate, secure the locking lugs, and aid in seating the rods in the lower tie plate throughout the life of the assembly. At EOL due to the combination of spring relaxation and differential rod growth between the tie rods and the other rods, the available spring force is reduced. However, since the spacer spring force is reduced at EOL, the force required to keep the rod in the lower tie plate is also reduced. The analysis shows that there is sufficient net holddown force to assure that both the fuel rods and water rods are firmly seated into the lower tie plate, to support the upper tie plate, and to secure the locking lugs.

3.16 Lower Tie Plate Seal

To provide constant positive contact between the seal and the channel as the channel deforms, the spring seal on each side of the lower tie plate has seven (7) individual leaf springs. The measured assembly growth for the demonstration assemblies at 38,030 MWD/MTU (see Section 2.1) is consistent with the SPC database on assembly growth (although slightly lower growth was measured than predicted by the average of the SPC database at that exposure). Using the SPC database to predict assembly growth and channel engagement leads to the conclusion that there will be no disengagement between the channel and seal spring at 45 GWD/MTU. This will be confirmed by the measurements to be performed on the demonstration assemblies which achieved an exposure of 46,848 MWD/MTU. Thus, the lower tie plate seal will function to limit leakage flow out to a bundle exposure of 45 GWD/MTU.

4.0 CONCLUSION

Currently, the NRC approved licensing mechanical design analysis limits the exposure on SPC 9X9-2 fuel assemblies to 40 GWD/MTU. Based on the fuel inspection results, the in-reactor demonstration, and the fuel mechanical design analyses presented in this report, an increase in the licensed fuel assembly limit to 45 GWD/MTU is justified.

5.0 REFERENCES

1) EMF-92-052(P), Rev. 1, "Susquehanna SES Mechanical Design Report For Extended Exposure of 9X9-2 Fuel Assemblies," Siemens Power Corporation - Nuclear Division, February 1994.

2) Letter from R. J. Clark (NRC) to H. W. Keiser (PP&L) "Cycle 6 Reload, Susquehanna Steam Electric Station, Unit 2 (TAC No. M83988)," October 28, 1992.

3) EMF-92-019(P), Rev. 0, "Inspection of SNP 9X9 BWR Fuel Assemblies After Three Cycles At Susquehanna Unit 2, Siemens Nuclear Power Corp., December 1991," March 1992.

4) XN-NF-85-67(P)(A), Rev. 1, "Generic Mechanical Design For Exxon Nuclear Jet Pump BWR Reload Fuel," September 1986.

5) EMF-92-191(P), Rev. 1, "Mechanical Design Analysis for the Susquehanna Power Uprate," Siemens Power Corp., April 1994.

6) XN-NF-82-06(P)(A) & Supplements 2, 4, & 5, Rev. 1, "Qualification of Exxon Nuclear Fuel for Extended Burnup," October 1986.

Figure 1

Revised Steady State
9X9-2 LHGR Limits

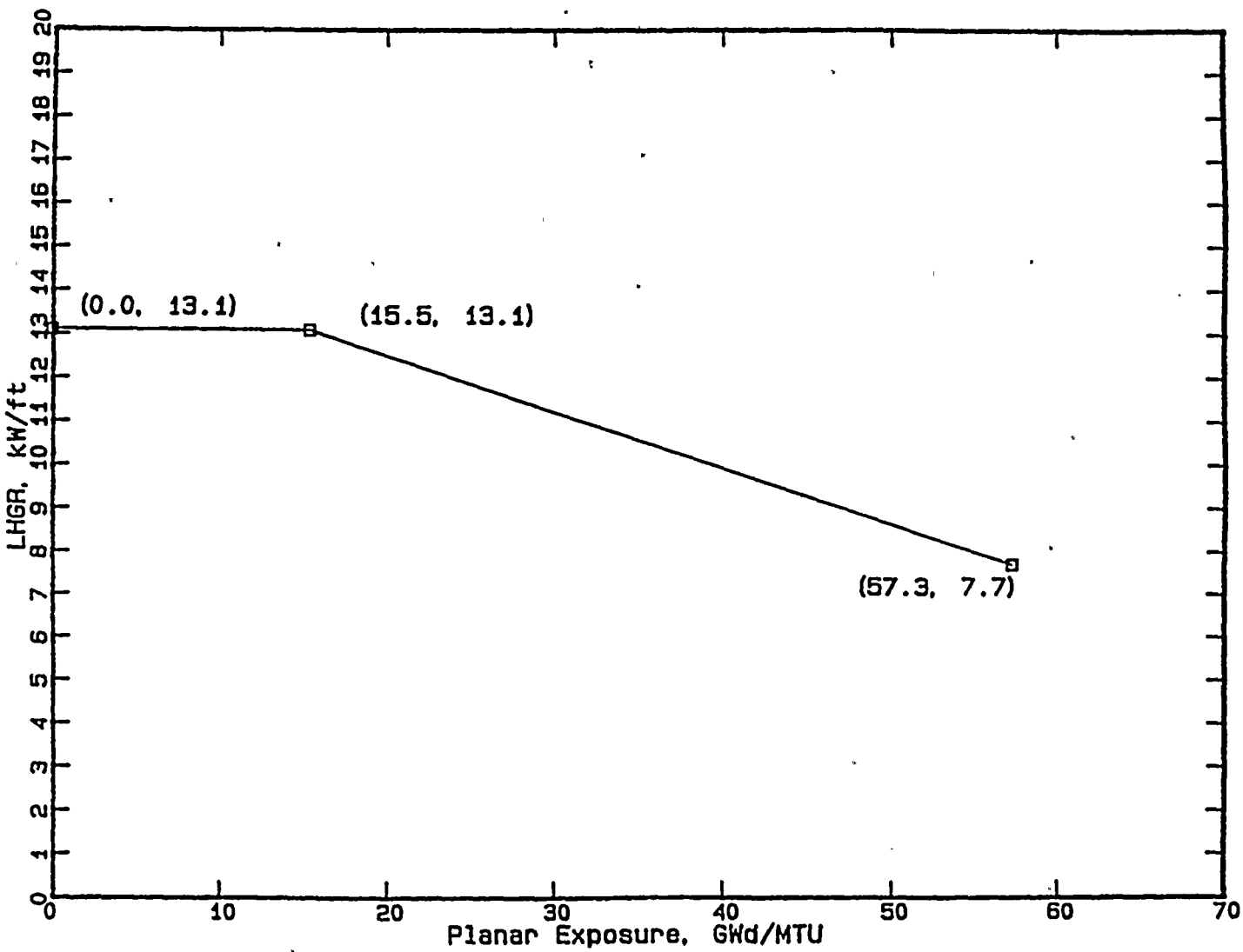

Figure 2

Revised 9X9-2 Limits for
Protection Against Power Transients