UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

Before the Atomic Safety and Licensing Board

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| In the Matter of | ) | |
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| LONG ISLAND LIGHTING COMPANY | ) | Docket No. 50-322-OL-4 |
| | ) | (Low Power) |
| (Shoreham Nuclear Power Station, | ) | |
| Unit 1) | ) | |

TESTIMONY OF
ATAMBIR S. RAO, EUGENE C. ECKERT,
GEORGE F. DAWE AND ROBERT M. KASCSAK

1. Q. Please state your names and business addresses.[1/]

A. (Rao) My name is Atambir S. Rao. My business address is General Electric Company, 175 Curtner Avenue, San Jose, California 95125.

A. (Eckert) My name is Eugene C. Eckert. My business address is General Electric Company, 175 Curtner Avenue, San Jose, California 95125.

A. (Dawe) My name is George F. Dawe. My business address is Stone & Webster Engineering Corporation, 245 Summer Street, Boston, Massachusetts 02107.

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1/ This testimony is jointly sponsored by all four witnesses; however, where appropriate, a lead witness has been designated.

A. (Kascsak) My name is Robert M. Kascsak. My business address is Long Island Lighting Company, Shoreham Nuclear Power Station, Wading River, New York 11792.

2. Q. Dr. Rao, what is your current position with the General Electric Company?

A. (Rao) My current position with General Electric, to which I was appointed on April 1, 1984, is Senior Program Manager, Advanced Engineering. Prior to that I was Manager, Plant Safety Systems Engineering.

3. Q. What were your responsibilities as Manager, Plant Safety Systems Engineering?

A. (Rao) My responsibilities included managing a group of engineers who performed a variety of plant safety performance analyses. The major emphasis was on determining the response of the core and containment following loss of coolant accidents (LOCA). Some of these evaluations were performed using state-of-the-art analysis models to determine the realistic response of the core and containment. These evaluations were in addition to FSAR safety analyses performed to show

compliance with regulations using conservative analytical models and assumptions. The group I managed included several engineers who have spent years doing such evaluations and are considered experts in their fields.

4. Q. Please summarize your prior professional and educational experience.

A. (Rao) I have previously held a number of positions relating to accident and transient analyses since I first joined General Electric in 1973. Earlier responsibilities have included modeling and analyzing the thermal hydraulic behavior of boiling water reactor (BWR) fuel following loss of coolant accidents, assessing the implication of advances in heat transfer, fluid mechanics, thermodynamics and two-phase flow on overall BWR system response during transients and loss of coolant accidents, developing emergency operator guidelines, and assessing containment thermal hydraulic and radiological response for various accidents and transients. At General Electric I have been assigned as Manager, Emergency Core Cooling Systems (ECCS) Engineering (1979-82), and Manager, Containment and Radiological Engineering

(1982-84). I received a Ph.D and a Masters Degree in Mechanical Engineering from the University of California, Berkeley, and a Bachelor of Technology in Mechanical Engineering from the Indian Institute of Technology, Kanpur, India. I am a member of the American Society of Mechanical Engineers (ASME) and a member of the Plant Systems Committee of the Nuclear Engineering Division of ASME.

5. Q. Mr. Eckert, what is your current position with the General Electric Company?

A. (Eckert) I am employed by the General Electric Company as Manager, Plant Transient Performance Engineering, a position I have held since 1971.

6. Q. What are your responsibilities in this position?

A. (Eckert) I am responsible for establishing the simulation requirements for the computer models needed to perform transient analyses, development of design procedures, evaluation of BWR stability, and evaluation and specification of the functional protection systems required for reactor abnormal transient protection in areas of overpressure, fuel thermal margin and related

plant performance. These responsibilities begin with initial BWR product line development, carry through the whole design period for specific power stations and extend through the initial startup testing and operating of each unit. This scope of responsibility has completely encompassed the development of the BWR/4 product line, the startup and operation of most of those units, and covered the entire progress of the Shoreham station.

7. Q. Please summarize your prior professional and educational experience.

A. (Eckert) Immediately upon joining General Electric Company in September 1959, I participated in assignments which included large jet engine control design, aircraft nuclear propulsion control analysis, nuclear submarine kinetics and control analysis, and industrial control simulation analysis at GE's Research and Development Center. In 1962, I joined General Electric's Nuclear Energy Division to work on boiling water reactor simulation and dynamic analysis. I have been responsible for design and licensing documentation of the dynamic analysis for several GE BWRs and have

participated in initial startup testing of many of the units. I received a Bachelor of Science Degree in Electrical Engineering from Valparaiso University in Indiana in 1958. I attended Stanford University under an Oak Ridge Fellowship and received a Master of Science Degree in Engineering Science in August 1959.

8. Q. Mr. Dawe, what is your current position with the Stone & Webster Engineering Company?

A. (Dawe) My current position, to which I was appointed in January 1980, is Supervisor of Project Licensing within the Licensing Division of Stone & Webster (SWEC).

9. Q. What are your responsibilities in this position?

A. (Dawe) I am responsible for technical and administrative supervision of all licensing personnel assigned to SWEC headquarters projects, including field assignments. My duties include assuring project awareness of regulatory requirements and developments, assuring proper and consistent application of SWEC licensing policies, and consulting with projects and clients on licensing related issues.

10. Q. Please summarize your prior professional and educational experience.

A. (Dawe) I joined Stone & Webster in 1973 as an Engineer in the Licensing Group. In January 1974, I was assigned as Licensing Engineer for the Shoreham Nuclear Power Station (SNPS) under construction, and was Lead Licensing Engineer from 1976 to 1980. In this capacity, I was responsible for all licensing related activities for SNPS including preparation of the Final Safety Analysis Report. This included the consolidation and review of Chapter 15, Accident Analysis. I have had additional assignments at Stone & Webster including development of company positions for NRC Regulatory Guides and Lead Licensing Engineer for the Special Projects Group of the Operations Services Division. I am also the Stone & Webster representative to, and participating member of, two subcommittees of the AIF Committee on Reactor Licensing and Safety.

Prior to joining Stone & Webster, I served seven years as a commissioned officer in the U.S. Navy Nuclear Power Program. My duties included direct supervision of operation and maintenance of a

submarine nuclear propulsion plant. I also served on the staff of the U.S. Navy Nuclear Power School as Director, Core Characteristics and Reactor Physics Division. While on active duty, I was qualified for assignment as Chief Engineer on nuclear powered vessels.

I received a Bachelor of Science degree from the United States Naval Academy in 1966. I have 18 years experience in the nuclear power field and hold a certificate as Engineer-in-Training in Massachusetts by 8 hour examination.

11. Q. Mr. Kascsak, what is your current position with LILCO?

A. (Kascsak) I am LILCO's Nuclear Systems Engineering Division Manager.

12. Q. What are your responsibilities in this position?

A. (Kascask) My responsibilities include overseeing an engineering staff organization capable of providing specialized engineering support associated with nuclear plant design. These capabilities include: systems engineering, thermal hydraulic analysis, materials engineering, reliability analysis and equipment

qualifications. This organization is also responsible for approving Architect/Engineer designs and vendor designs and, more recently, developing an in-house organization to support future plant modifications. My division participated in and reviewed the safety analysis for the Shoreham plant and is responsible for any modifications or additions to the Shoreham plant and its safety analysis.

13. Q. Please summarize your prior professional and educational experience.

A. (Kascsak) I joined LILCO in 1969 as an Assistant Engineer in the Mechanical and Civil Engineering Department. I worked on various fossil fuel power station projects in the capacity of Associate and Senior Engineer, including the Northport Power Station Unit 3 and Unit 4 mechanical engineering designs. From July 1974 to March 1975, I served as LILCO Lead Mechanical Engineer for Shoreham and for the Jamesport Nuclear Power Station. In March 1975, I joined the Shoreham Project Group as an Assistant Project Engineer, after which I assumed the responsibilities of Project Engineer. From March 1975 to January 1979, I was

Project Engineer for Shoreham. In this position I was responsible for the review and approval of design activities prepared by our Architect/Engineer, Nuclear Steam Supply System Vendor and LILCO in-house engineering departments. In January, 1979, I was appointed Manager of the Nuclear Systems Division.

I am a registered Professional Engineer in New York and a member of the American Society of Mechanical Engineers.

I graduated from Manhattan College in 1969 with a Bachelor of Mechanical Engineering. In 1977 I received a Master of Science degree in Nuclear Engineering from Polytechnic Institute of New York. I have completed training courses in BWR and PWR technology.

14. Q. Gentlemen, what is the purpose of your testimony?

A. At the request of the Long Island Lighting Company, General Electric, in conjunction with LILCO and Stone & Webster, reviewed all of the events considered in Chapter 15 of the FSAR to compare the effects on public health and safety of operation of the Shoreham plant during fuel load,

cold criticality testing and low power operations to the effects at or following full power operation. Each of us participated in this review and our testimony describes its results, including plant response to various transient and accident scenarios and time available for operator response in restoring AC power if necessary.

15. Q. Please describe what you mean by the Chapter 15 accident and transient analysis.

A. Chapter 15 of the Shoreham FSAR provides the results of analyses for the spectrum of accident and transient (anticipated operational occurrences) events that must be accommodated by the Shoreham plant to demonstrate compliance with the NRC's regulations. The results of the safety analysis demonstrate the ability of the plant to erate without undue risk to the health and fety of the public, even in the event of such cidents and transients. The adequacy of the pe of the Chapter 15 accident and transients ɔ demonstrate compliance with NRC regulations was the subject of previous hearings for Shoreham. The Licensing Board, in its Partial Initial Decision (PID) of September 21, 1983,

found that the spectrum of events analyzed in Chapter 15 is appropriate, adequate and not unique to Shoreham. PID at 180-84.

16. Q. Has the NRC reviewed and approved the accident and transient analysis contained in Chapter 15 of the Shoreham FSAR?

A. Yes. The Shoreham FSAR was submitted to the NRC Staff for review and has been approved by the Staff in its Safety Evaluation Report for Shoreham (NUREG-0420).

17. Q. Please describe your review of the Chapter 15 analyses in light of LILCO's proposal to operate the plant at low power levels prior to completion of hearings concerning the existing TDI diesel generators.

A. Although the FSAR considers all phases of operation of the plant from fuel load to operation at 100% power, this review was performed specifically to determine whether operation of the Shoreham plant during low pow r operation will pose undue risk to public health and safety given an assumed unavailability of the TDI diesels. The review of Chapter 15 was divided into three

parts: fuel load and precriticality testing (Phase I), cold criticality testing (Phase II) and low power testing up to 5% of rated power (Phases III and IV). Although Phases III and IV were considered together, operation during Phase III presents even less risk than the low risk associated with operation during Phase IV. The review was based upon the same criteria and bases as the original Chapter 15 analyses.

18. Q. Explain how this review differed from the original Chapter 15 analysis performed for full power operation?

A. For the large majority of events addressed in Chapter 15, there is no change. Review shows that either the event cannot occur, or assumptions concerning loss of AC power are neither appropriate nor required. Thus, if the event can occur, it is clearly bounded by the full power analysis already presented in Chapter 15 with acceptable results. For the few events where additional calculations were required, as described later in this testimony, we specifically considered plant operating histories for power levels up to 5% of full power to establish fission

product inventories and decay heat rates. Further, we performed the analyses assuming the TDI diesels were not available. Thus, where loss of offsite power was postulated, we assumed the plant would have no AC power until some source could be restored.

19. Q. What are the results of your review?

A. The review of Chapter 15 demonstrates that operation during fuel load and low power testing proposed by LILCO will not result in any undue risk to the public health and safety. In fact, the risk from any Chapter 15 event during both the fuel load and precriticality phase (Phase I) and the cold criticality testing phase (Phase II) is non-existent. The risk to the public health and safety from the Chapter 15 events postulated for low power testing up to 5% of rated power (Phases III and IV) is smaller than the risks already found acceptable for 100% power operation. As already indicated, this review considered the TDI diesels to be unavailable for the sake of analysis.

Phase I: Fuel Loading and Precriticality Testing

20. Q. Please describe the activities to be conducted during Phase I of low power testing program.

A. This phase of operation of the Shoreham plant includes only initial fuel loading and precriticality testing. The reactor will remain at essentially ambient temperature and atmospheric pressure. The reactor will not be taken critical. Any increase in temperature beyond ambient conditions will be due only to heat sources external to the core such as recirculation pump heat. There will be no heat generation in the core. Details of the steps to be performed during these operations are described in the testimony of by Mr. William E. Gunther.

21. Q. What does your Chapter 15 review for Phase I lead you to conclude?

A. Of the 38 accident or transient events addressed in Chapter 15, 18 of the events could not occur during Phase I because of the operating conditions of the plant. An additional six events could physically occur, but given the plant conditions, would not cause the phenomena of

interest in the sense of the Chapter 15 safety analysis. The remaining 14 events could possibly occur, although occurrence is highly unlikely given the plant conditions. In any event, and most importantly from a safety standpoint, the potential consequences of these 14 events would be trivial. Attachment 1 is a summary of our review for Phase I which lists the category into which each Chapter 15 event falls.

22. Q. Which of the Chapter 15 events could not occur during Phase I?

A. There are 18 Chapter 15 events which could not occur during Phase I. These events are identified in Attachment 1.

23. Q. Why can't these events occur?

A. The 18 Chapter 15 events which could not occur during Phase I are precluded by the operating conditions of the reactor. These events all involve operating modes or component operation which are not possible during this phase. For example, during fuel loading and precriticality testing, the reactor is at essentially ambient temperature and atmospheric pressure.

Accordingly, no steam is available. Thus, all events which occur only under pressurized conditions are impossible. These include the following events:

1. generator load rejection
2. turbine trip
3. turbine trip with failure of generator breakers to open
4. MSIV closure
5. pressure regulator failure - open
6. pressure regulator failure - closed
8. loss of feedwater heating
10. inadvertent HPCI pump start
17. inadvertent opening of a S/R valve
21. loss of condenser vacuum
31. main condenser gas treatment system failure
35. pipe breaks outside primary containment (steam line break accident)
38. failure of air ejector lines

In addition, the following events are precluded by definition:

11. continuous control rod withdrawal during power range operation
15. off-design operation transients due to inadvertent loading of a fuel assembly into an improper location
16. inadvertent loading and operation of a fuel assembly in improper location
27. ATWS
28. cask drop accident

24. Q. In addition to the 18 events that cannot occur,

you indicated that six events could physically occur but that the resultant phenomena of interest in the Chapter 15 analysis could not exist. Please explain this conclusion for each of those six events.

A. These events are:

20. recirculation pump trip
22. recirculation pump seizure
23. recirculation pump flow control failure - decreasing flow
24. recirculation flow control failure - with increasing flow
25. abnormal startup of idle recirculation pump
26. core coolant temperature increase

All five recirculation pump events would be of interest only if they could affect core physics or thermal-hydraulic conditions thus changing core criticality, heat transfer or both. With the core subcritical and no heat generation or boiling in the core, there are no pertinent phenomena (such as temperature differences or void collapses) to evaluate. Another event, the core coolant temperature increase event, postulates a loss of residual heat removal (RHR) system cooling. Even if the RHR system were operated in Phase I, without decay heat there would be no

temperature increase to evaluate should the RHR system be lost.

25. Q. What Chapter 15 events could occur during Phase I?

A. The remaining 14 Chapter 15 events which could possibly occur are identified in Attachment 1.

26. Q. Could any of these events have adversely impacted public health and safety?

A. No. All are trivial events with no potential to impact public health and safety adversely.

27. Q. Could you explain your conclusions?

A. These 14 events can be divided into reactor events and non-reactor events. The reactor events, those which involve some response of the reactor, are:

7. feedwater controller failure - maximum demand
9. shutdown cooling (RHR) malfunction - decreasing temperature
12. continuous rod withdrawal during reactor startup
13. control rod removal error during refueling
14. fuel assembly insertion error during refueling
18. loss of feedwater flow
19. loss of AC power
30. off design operational transient as a

consequence of
instrument line failure
33. control rod drop accident
34. pipe breaks inside the
primary containment
(loss of coolant accident)
37. feedwater system piping break

Since the core contains no fission product inventory and therefore no decay heat, these events cannot induce a transient that could challenge the fuel and therefore do not require any mitigating action. For the non-reactor events, which include miscellaneous small releases outside primary containment (event 29), liquid radwaste tank rupture (event 32) and fuel handling accident (event 36), no radiological consequences would result since no fission products exist. Therefore, they would pose no risk to public health and safety.

28. Q. Does a loss of coolant accident present any risk during Phase I operations?

A. No. A loss of coolant accident (event 34) would have no consequences during Phase I since no core cooling is required. No fission products exist and therefore no decay heat is available to heat up the core. The fuel simply would not be challenged even by a complete draindown of the reactor vessel for an unlimited period of time.

29. Q. In your opinion, is there any possible risk to the public health and safety during Phase I operations?

A. No. Many Chapter 15 events simply cannot occur, and for those that can, there can be no consequences adversely affecting the health and safety of the public.

30. Q. Does your conclusion that there will be no risk to public health and safety in Phase I depend in any way on the availability of the TDI diesel generators at Shoreham?

A. No. This conclusion is not affected by any postulated diesel generator unavailability because it is in no way dependent on the availability of any AC power. Simply, no AC power is necessary during Phase I to cool the core.

<u>Phase II: Cold Criticality Testing</u>

31. Q. Please describe the activities to be conducted during cold criticality testing.

A. This phase of low power testing of the Shoreham plant will include cold criticality testing of the plant at essentially ambient temperature and

atmospheric pressure. The power level during this phase of testing will be in the range of .0001% to .001% of rated power. Details of the testing to be performed during this phase are described in the testimony of William Gunther.

32. Q. Please state the conclusions resulting from your review of the Chapter 15 analysis during Phase II.

A. Fifteen of the events could not occur during Phase II because of the operating conditions of the plant. The remaining 23 events could occur but each event leads to essentially no consequences. This analysis is summarized in Attachment 2.

33. Q. What events cannot occur during Phase II operations?

A. The following 15 Chapter 15 events cannot occur during Phase II:

1. generator load rejection
2. turbine trip
3. turbine trip with failure of generator breakers to open
4. MSIV closure
5. pressure regulator failure - open
6. pressure regulator failure - closed
8. loss of feedwater heating

10. inadvertent HPCI pump start
11. continuous control rod withdrawal during power range operation
17. inadvertent opening of a safety/relief valve
21. loss of condenser vacuum
28. cask drop accident
31. main condenser gas treatment system failure
35. pipe breaks outside primary containment (steam line break accident)
38. failure of air ejector lines

34. Q. Please explain why these 15 events cannot occur.

A. Thirteen of these events cannot occur because the reactor will be at essentially ambient temperature and atmospheric pressure and no steam will be generated. This is true except for event 11, continuous control rod withdrawal during power range operation and event 28, cask drop accident. Event 11 is precluded because the event is defined only for power range operation which is never achieved in Phase II, and event 28, as noted in FSAR § 15.1.28, is precluded by plant design.

35. Q. Of the remaining 23 events that can occur, are they less likely to occur during Phase II operation than during normal operations? Please explain.

A. Many of the 23 events remaining in the Chapter 15 analysis are far less likely to occur during Phase II testing than during normal operations. For example, the recirculation pump trip (event 20), the recirculation pump seizure (event 22), the recirculation flow control failures (events 23 and 24) and the abnormal startup of idle recirculation pump (event 25), although physically possible, are not as likely to occur because the recirculation pumps are used for only limited periods of time during this phase of the testing program. Similarly, the loss of feedwater event (event 18) is very unlikely because, without boil-off, little, if any, make-up water will have to be supplied to the reactor. Moreover, make-up water would not normally be supplied by the feedwater system given the low flow requirements under these conditions. Miscellaneous small releases outside primary containment (event 29) are very unlikely due to the minimal amount of radioactive material in the plant. An off design operational transient as a consequence of instrument line failure (event 30) or a feedwater system piping break (event 37) is very unlikely given the unpressurized condition

of the plant. Thus, many of the Chapter 15 events that are physically possible during Phase II remain very unlikely in light of the plant conditions that will then exist.

36. Q. Although less likely to occur, did you evaluate the consequences of the 23 possible events identified in Attachment 2?

A. Yes. All 23 possible events contained in Chapter 15 were reviewed to reaffirm that the consequences of these events, should one occur during Phase II of low power testing, would be bounded by the consequences of the event presented in the FSAR for full power operation.

37. Q. How did you reach your conclusion that all events are bounded by FSAR analysis?

A. Each event was evaluated at the operating conditions in Phase II and the consequences of the event considered.

For example, the continuous control rod withdrawal during startup event (event 12) can be postulated to occur in the power, source and/or intermediate range of operation. During cold functional criticality testing, the reactor will

operate in the source and intermediate ranges and therefore the conclusions contained in Chapter 15 are applicable to this event should it occur during this phase of low power testing. As the FSAR indicates, this event would not result in any release of radioactive material from the fuel at any power level.

Another example is the fuel handling accident (event 36). As stated in the FSAR, the most severe fuel handling accident is a dropping of a fuel assembly onto the top of the core. The FSAR analysis assumes that the fuel contains a fission product inventory equivalent to operation of 1000 days at full rated power. This assumption results in an equilibrium fission product concentration at the time the reactor is shut down. In other words, longer operating histories could not increase the concentration of the fission products of concern. But as already noted, the fission product inventories in the core will be significantly less during Phase II low power testing than the inventories analyzed in the FSAR because of the extremely low power levels (.0001% to .001% of rated power) achieved during this testing and the limited time at those power

levels. Even if a fuel handling accident took place and fuel damage did occur, there would be significantly fewer fission products to be released from the core than those that have already been analyzed and found acceptable in the FSAR. In addition, no fuel handling is required in Phase II and, thus, a fuel handling accident is unlikely.

A third example is the liquid radwaste tank rupture event (event 32). This event assumes the rupture of a radwaste tank that contains a substantial amount of contaminated liquids generated during the operation of the reactor. But again, since Phase II of low power testing results in insignificant power levels in the reactor, there will be little, if any, radioactive liquids generated and collected in the radwaste tank. Consequently, a rupture of the tank would have consequences far less severe than those found acceptable in Chapter 15.

38. Q. Did your review of Chapter 15 events for Phase II consider the unavailability of the Shoreham TDI diesels?

A. Yes. The review of Chapter 15 events for Phase II testing and the conclusions reached are unaffected by unavailability of the TDI diesels. Of the 23 possible Chapter 15 events reviewed, 20 do not require the assumption of loss or unavailability of offsite AC power. See Attachment 2. Therefore, our conclusions for these 20 of the 23 possible events are independent of the status of the diesels.

The three events that do assume loss or unavailability of offsite AC power are (1) pipe breaks inside the primary containment (LOCA) (event 34), (2) feedwater system piping break (event 37) and (3) the loss of AC power event (event 19).

39. Q. For the events that assume the loss of offsite AC power, did you also consider the unavailability of the TDI diesels?

A. Yes. With respect to these events, the LOCA (pipe break inside containment, event 34) would be the most limiting event and bounds the other two events. That is, the LOCA conditions present a greater challenge to the fuel than either the feedwater system piping break or the loss of AC

power, neither of which affect the reactor coolant inventory during Phase II. If a LOCA did occur during the cold criticality testing phase, however remote that possibility, there would be time on the order of months available to restore make-up water for core cooling. At the power levels achieved during Phase II, fission product inventory is very low. At most, the average power output will be a fraction of a watt per rod, with no single rod exceeding approximately 2 watts. This is less, roughly, than the heat output of a Christmas tree bulb. With these low decay heat levels, the fuel cladding temperature would not exceed the limits of 10 CFR § 50.46 even after months without restoring coolant and without any source of AC power. Thus, there is no need to rely on the TDI diesel generators, or any source of AC power.

For the non-LOCA events, the loss of AC power event (event 19) and the feedwater system piping break (event 37), during cold criticality testing conditions there is similarly no need to rely on the diesel generators for mitigation of the event. For these events, no loss of coolant occurs and the decay heat is minimal. Thus, core

cooling can be achieved for unlimited periods of time without AC power using the existing core water inventory and heat losses to ambient.

40. Q. Are there any other reasons to conclude that the three events that assume loss of AC power present less risk during Phase II than during normal operation?

A. Yes, the LOCA (event 34) and the feedwater system piping break (event 37) postulate the double ended rupture of a piping system. Because the reactor will be at essentially ambient temperature and atmospheric pressure during Phase II, it is extremely unlikely that such a pipe break would ever occur. In fact, the NRC Staff does not require double ended ruptures to be postulated for low temperature and low pressure systems in safety analyses. Thus, these events are much less likely during cold criticality testing than during normal operation. Even during normal operation, these are highly unlikely events.

41. Q. Can you summarize your conclusions relative to the consequences of the possible Chapter 15 events that could occur during Phase II?

A. As we have demonstrated above, the acceptable consequences already described in the FSAR for the design basis events would be further reduced under these low power testing conditions. For each of these events, the consequences are significantly less severe for any event occurring during the cold functional criticality testing than for the event analyzed in Chapter 15. Because of the extremely low power levels reached during this testing phase, fission product inventory in the core will be only a small fraction of that assumed for the Chapter 15 analyses. As indicated above, the FSAR assumes operation at 100% power for 1000 days in calculating fission product inventory; the inventory during Phase II low power testing will be less than one one-hundred-thousandth (.00001) of the fission product inventory assumed in the FSAR. Consequently, none of the events analyzed in Chapter 15 could result in a release of radioactivity during cold criticality testing that would endanger the public health and safety.

42. Q. Do your conclusions depend in any way on the availability of the TDI diesels?

A. No. In fact, even if AC power were not available for extended periods of time, fuel design limits and design conditions of the reactor coolant pressure boundary would not be approached or exceeded as a result of anticipated operational occurrences, and the core would be adequately cooled in the unlikely event of a postuated accident.

Phases III and IV: Low Power Testing Up to 5% of Rated Power

43. Q. Please describe the activities that will be conducted during Phases III and IV of low power testing.

A. Phase III low power testing will include operation of the plant at power levels up to 1% of rated power while Phase IV will include operation at power levels from 1 to 5% of rated power. Details of the testing to be performed during each of these phases are described in William Gunther's testimony.

44. Q. Is it necessary to consider Phases III and IV together or can each of these phases be considered separately?

A. For evaluation purposes, these phases can be considered together for convenience. They are, however, distinct phases of operation. The results do differ for the two phases in that more time is available to restore AC power following initiation of the limiting event (LOCA) from the Phase III power level (1%) than from the Phase IV power level (5%).

45. Q. What resulted from your review of Chapter 15 for Phases III and IV?

A. Operation testing at power levels up to 5% of rated power poses no undue risk to the public health and safety. In fact, risk is substantially less than that already found to be acceptable by the NRC Staff in its review of Chapter 15.

Even if the Shoreham TDI diesels are assumed to be unavailable, there is ample assurance that fuel design limits and design conditions of the reactor coolant pressure boundary will not be exceeded as a result of anticipated operational occurrences, and that the core will be cooled and containment integrity and other vital functions will be maintained in the event of any postulated accident. Analyses for the bounding event (the

LOCA), using very conservative regulatory assumptions demonstrate calculated operator action times of 370 minutes and 86 minutes for Phases III and IV, respectively. Calculations based on more realistic assumptions result in calculated operator action times of greater than 24 hours and three hours for Phases III and IV, respectively.

46. Q. Can all 38 Chapter 15 events occur during Phases III and IV?

A. No. Two of the events in Chapter 15, generator load rejection (event 1) and turbine trip with generator breaker failure (event 2) cannot occur because the generator will not be connected to the grid during these phases of testing. A third event, the cask drop (event 18), is precluded by design as stated in the FSAR. See Attachment 3.

47. Q. Of the 35 events that could conceivably occur, how do the conclusions of your review of operation during Phases III and IV compare to those from the Chapter 15 analyses for full power?

A. Operation of Shoreham during Phases III and IV will result in substantially reduced risk because

1) during these phases, reactor power will remain below 5% of rated power, thereby reducing the potential consequences of an event;

2) there is more time available for preventive or mitigating action than would be the case for the events as analyzed in Chapter 15; and

3) the required capacity for mitigation systems is less than that required for the Chapter 15 analysis.

48. Q. The first factor contributing to the significantly lower risk during low power operation is the power limitations to be imposed during these phases. How does this reduce risk?

A. With power limited during low power testing to 5% of rated power, the fission product inventory in the core will not exceed 5% of the equilibrium values assumed in the FSAR. In fact, because of the intermittant type of operations conducted during low power testing, equilibrium fission product inventory for even 5% power will not actually be achieved. This reduced fission product inventory reduces risk in two ways: first, the amount of decay heat present in the core following shutdown is substantially reduced, and second, the amount of radioactivity that could be

released should a fuel failure occur is substantially reduced.

49. Q. The second factor contributing to the significantly lower risk during low power operations is the increased time available for preventive or mitigating action. Please explain how this reduces the risk during low power operation.

A. Longer time is available because the limited power levels result in longer times for the plant to reach setpoints and limits. For example, on loss of feedwater (event 18), the water level in the reactor will decrease at a slower rate due to the reduced steaming rate than if the event occured at 100% power. This gives the operator more time to act manually to restore feedwater before an automatic action takes place. Similarly, in the loss of condenser vacuum event (event 21), the operator may have more time to identify the decreasing vacuum and to take steps to remedy the situation before automatic actions such as turbine trip, feedpump trip or main steam isolation occur. Another example is the main steam isolation valve closure event (event 4). At five percent power, the amount of heat

produced upon isolation of the reactor vessel (which is followed by a reactor scram) results in a much slower pressure and temperature increase than would be experienced at 100% power. This gives the operator more time to manually initiate reactor makeup rather than relying on automatic action. In short, the operator may end the transient before setpoints for automatic protection actions are reached. The setpoints themselves are established with margin to any limit which could result in substantial impact on the plant.

50. Q. With respect to the third factor, reduction in required capacity for mitigating systems, please explain how this reduces the risk during low power testing.

A. Since there are lower levels of decay heat present following operation at 5% power, the demand for core cooling and auxiliary systems is substantially reduced, permitting the operation of fewer systems and components to mitigate any event. It follows that the AC power requirements for event mitigation are substantially reduced for 5% power operation as compared to 100% power operation.

51. Q. How many of the Chapter 15 event that are possible during Phases III and IV require the assumption of the unavailability of off-site power?

A. Four. The events are: loss of AC power (event 19), pipe breaks inside the primary containment (loss of coolant accident) (event 34), pipe breaks outside primary containment (steam line break accident) (event 35), and feedwater system piping break (event 37).

52. Q. Of these four events, what is the potentially most significant event?

A. The pipe break inside the primary containment (loss of coolant accident) event (event 34) is potentially the most significant because it has the potential for the most severe loss of coolant inventory in the reactor vessel.

53. Q. In the unlikely event that a loss of coolant accident occurs during Phase III, how much time is available to restore core cooling?

A. (Rao) Greater than 24 hours. This conclusion is the result of calculations, based on 10 CFR 50 Appendix K models with some realistic assumptions, which show that the peak cladding

temperature after 24 hours of no coolant make-up is less than 1600F. Hence core cooling does not have to be restored for some time in excess of 24 hours.

Even calculations using the very conservative assumptions required by 10 CFR 50.46 and Appendix K demonstrate that the operator has greater than 370 minutes before core cooling has to be restored to prevent the core from exceeding 10 CFR 50.46 limits for peak cladding temperature, local oxidation or core wide oxidation.

54. Q. You mentioned two different analyses, an analysis using some realistic assumptions and a very conservative Appendix K analysis. Please explain the pertinent assumptions made in each of these analyses.

A. (Rao) Let me start with the Appendix K analysis. This conservative analysis used all the same models that are required by 10 CFR 50.46 and Appendix K, and which are used in the FSAR Chapter 15 analysis for 100% power operation. These models have been approved by the NRC Staff. Among other conservatisms, this Appendix K analysis assumed that during Phase III the decay heat would be

approximately 1% of the value used in the FSAR analysis for full power operation. This assumption is conservative because it would take a substantial amount of time operating continuously at 1% power to accumulate this level of core fission product inventory. During Phase III testing, extended operations at 1% power will not occur. A second significant conservative assumption is that no convective heat transfer occurs following the initial blowdown, and loss of inventory until spray or injection is initiated. In reality, of course, even with no cooling water present in the core, heat transfer to the surroundings occurs. Given the low decay heat levels present in the core, this heat transfer is a significant factor in determining when cooling water needs to be restored.

The more realistic analysis was performed by removing some, but not all, of the major conservatisms that exist in the Appendix K analysis. In particular, conservatisms relating to the decay heat and natural convection heat transfer were treated more realistically. The decay heat inventory used in this calculation was based on the latest American Nuclear Society Standard (ANS

5.1, "Decay Energy Release Rate Following Shutdown of Uranium-Fueled Thermal Reactors") and took into account the actual anticipated operating history of the core during Phase III. This calculation was also based on a core power peaking factor of 1.38 which results from the control rod withdrawal pattern which is planned for use at Shoreham during this phase of testing. This calculation also considers the effects of the natural convection heat transfer. This is the heat transferred from the rods to the surrounding air and/or steam in the reactor following a LOCA. The amount of heat transfer that will occur in a dry core was determined based on natural convection flows between the core and the bypass regions.

55. Q. What effect does the use of ANS 5.1 have on your calculation of decay heat?

A. (Rao) The ANS 5.1 incorporates the results of additional research performed since the development of the 10 CFR 50.46 and Appendix K referenced model. The use of this standard results in a lower calculated power following reactor shutdown. This consequently results in the longer calculated times for operator action.

56. Q. How does the control rod withdrawal pattern affect your calculations?

A. (Rao) The control rod withdrawal pattern used in the calculation affects the local power in the various bundles. The pattern determines the peak power, expressed by the term peaking factor, in the bundle. Our calculations for the amount of time available to restore core cooling have been made for the highest power rod. Thus, even though the peak rod reaches the specified limits the bulk of the core is below the 10 CFR §50.46 limits.

57. Q. Why is it appropriate to use a core peaking factor of 3.38?

A. (Rao) This core peaking factor is appropriate because it is based on the planned control rod pattern developed for low power operation and planned for use during this phase of testing at Shoreham.

58. Q. You indicate that it is realistic to conclude that at least 24 hours are available before the plant would exceed the limits of 10 CFR § 50.46. Would any fuel failure occur prior to that time?

A. (Rao) No fuel failures (cladding perforations) would occur. In both the realistic and conservative Appendix K analyses there is significant margin between the calculated cladding temperature and the temperature at which cladding perforations would occur.

59. Q. During Phase IV, how much time is available to restore core cooling in the event of a LOCA?

A. (Rao) Greater than three hours. This conclusion is based on a calculation, using 10 CFR Part 20 Appendix K models with some realistic assumptions, which shows that the operator has greater than three hours before core cooling has to be restored to prevent the core from exceeding 10 CFR § 50.46 limits.

Even calculations using the very conservative assumptions required by 10 CFR § 50.46 and Appendix K show that the operator has greater than 86 minutes before core cooling has to be restored to prevent exceeding 10 CFR § 50.46 limits. This calculation is based upon the planned control rod pattern which results in a core power peaking factor of 3.38.

The Sherwood Affidavit submitted with LILCO's supplemental low power motion on March 20, 1984, stated that it would take 55 minutes to exceed 10 CFR § 50.46 limits. Sherwood Affidavit, Exhibit 4. That analysis used an extremely conservative peaking factor of 5, however. The peaking factor of 3.38 noted above is appropriate because the control rod pattern established for use during low power testing at Shoreham will limit the peaking factor to this value.

An additional calculation was performed using Appendix K assumptions with the anticipated power history of the core (i.e., 5% power for no greater than 60 days instead of 5% power for an infinite time). The calculated operator action time is 110 minutes with this additional assumption.

60. Q. Please explain the assumptions with respect to each of the scenarios you just described.

A. (Rao) The assumptions made for the more realistic analysis and the conservative Appendix K analysis for Phase IV are the same as those made for Phase III and described above. In addition, as explained earlier, we performed a modified

Appendix K calculation which reflects the actual operating history expected during Phase IV.

61. Q. The times specified for Phase IV are the times available prior to exceeding the limits of 10 CFR § 50.4.. Would any fuel perforations or core damage occur prior to exceeding the limits of 10 CFR § 50.46?

A. (Rao) No fuel perforations or core damage occurs since, in either the conservative or realistic analyses, the peak cladding temperature of the fuel will remain significantly below the cladding perforation temperature.

62. Q. In addition to the loss of coolant accident, you indicated that three other events require the assumption of loss or unavailability of offsite power. Assuming all AC power is lost, would core cooling be maintained for these events?

A. For these events the reactor would automatically isolate and both HPCI and RCIC would be available to provide reactor coolant makeup. Either of these systems have adequate coolant makeup capability to provide any required core cooling. In these events, reactor isolation, scram and

operation of HPCI and RCIC are independent of AC power.

63. Q. What if loss of offsite power is induced by a seismic event?

A. It is unlikely Shoreham would experience a seismic event causing loss of offsite power. However, as stated previously, response to a loss of offsite power is not dependent upon the availability of AC power. The HPCI and RCIC systems are seismically qualified and would operate automatically to ensure core cooling. These systems are steam driven and utilize DC power supplies.

64. Q. How long can the DC power supplies operate?

A. (Kascsak) A minimum of 24 hours. Evaluations performed for a "station blackout" with the plant operating at 100% power indicate that these power supplies (i.e., batteries) will operate for 24 hours and probably longer.

65. Q. What if AC power is not restored in the 24 hour time frame?

A. (Eckert, Kascsak) There is no immediate danger to the core. Even if DC power were lost after 24

hours, and AC power had not been restored, sufficient core cooling would be provided for at least 2 more days by available vessel inventory. Moreover, the battery chargers for the DC batteries can be supplied by a small portable generator set that is available on site. Thus, DC power availability would be retained beyond the initial 24 hours even if AC power were not restored.

66. Q. Assuming availability of DC power supplies, how much time is available for restoration of AC power?

A. (Rao) Since the availability of DC power ensures the operation of RCIC or HPCI, core cooling would be assured for indefinite periods of time. The only need for the restoration of AC power would be for containment cooling. The containment and suppression pool limits would not be exceeded for a approximately 30 days without AC power, given the heat capacity of passive heat sinks such as steel and concrete. Therefore, there is ample time for AC power to be restored.

67. Q. Are there any other reasons for concluding that these three events present less risk during low power operations than during normal operations?

A. (Dawe) Assuming the loss of offsite power in the context of pipe breaks outside containment (main steam line break accident and feedwater system break accident) is a conservatism which stems from the methodology for pipe breaks outside containment. That methodology requires the assumption of a loss of offsite power for pipe breaks which result directly in a plant trip of the turbine generator system or reactor protection system. Notwithstanding grid stability analyses which show otherwise, it is conservatively assumed that plant trips would cause perturbations of the grid, resulting in the loss of offsite power. For operation at 5% power or less, however, the turbine generator is not connected to the grid, and therefore any assumption of induced perturbation to the offsite grid is even more conservative.

68. Q. Please summarize your conclusions with respect to your review of Chapter 15 for Phases III and IV.

A. Operation of the plant during low power testing up to levels of 5% of rated power poses no undue risk to the public health and safety. In fact, any risk is substantially less than that already

found to be acceptable by the NRC Staff in its review of Chapter 15. Even if the Shoreham TDI diesels are assumed to be unavailable, there is ample assurance that fuel design limits and design conditions of the reactor coolant pressure boundary will not be exceeded as a result of anticipated operational occurrences, and that the core will be cooled and containment integrity and other vital functions will be maintained in the event of any postulated accident.

Analyses for the bounding event (the LOCA) show that, even using the very conservative regulatory assumptions, the calculated action times are 370 minutes and 86 minutes for Phases III and IV, respectively. More realistic calculations result in calculated operator action times of greater than 24 hours and three hours for Phases III and IV, respectively.

69. Q. Do any of these conclusions depend upon the availability of the TDI diesels?

A. No.

Attachment 1

## FUEL LOAD AND PRECRITICALITY TESTING

| Chapter 15 Event | Event Category |
|---|---|
| 1. Generator Load Rejection | * |
| 2. Turbine Trip | * |
| 3. Turbine Trip with Failure of Generator Breakers to Open | * |
| 4. MSIV Closure | * |
| 5. Pressure Regulator Failure - Open | * |
| 6. Pressure Regulator Failure - Closed | * |
| 7. Feedwater Controller Failure - Maximum Demand | *** |
| 8. Loss of Feedwater Heating | * |
| 9. Shutdown Cooling (RHR) Malfunction - Decreasing Temperature | *** |
| 10. Inadvertent HPCI Pump Start | * |
| 11. Continuous Control Rod Withdrawal During Power Range Operation | * |

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* Event not possible.

** Component operation possible but Chapter 15 phenomena cannot occur.

*** Event possible but no consequences.

| | | |
|---|---|---|
| 12. | Continuous Rod Withdrawal During Reactor Startup | *** |
| 13. | Control Rod Removal Error During Refueling | *** |
| 14. | Fuel Assembly Insertion Error During Refueling | *** |
| 15. | Off-Design Operational Transients Due to Inadvertent Loading of a Fuel Assembly into an Improper Location | * |
| 16. | Inadvertent Loading and Operation of a Fuel Assembly in Improper Location | * |
| 17. | Inadvertent Opening of a Safety/Relief Valve | * |
| 18. | Loss of Feedwater Flow | *** |
| 19. | Loss of AC Power | *** |
| 20. | Recirculation Pump Trip | ** |
| 21. | Loss of Condenser Vacuum | * |
| 22. | Recirculation Pump Seizure | ** |
| 23. | Recirculation Flow Control Failure - Decreasing Flow | ** |
| 24. | Recirculation Flow Control Failure With Increasing Flow | ** |
| 25. | Abnormal Startup of Idle Recirculation Pump | ** |
| 26. | Core Coolant Temperature Increase | ** |
| 27. | Anticipated Transients Without Scram (ATWS) | * |
| 28. | Cask Drop Accident | * |
| 29. | Miscellaneous Small Releases Outside Primary Containment | *** |

| | | |
|---|---|---|
| 30. | Off Design Operational Transient as a Consequence of Instrument Line Failure | *** |
| 31. | Main Condenser Gas Treatment System Failure | * |
| 32. | Liquid Radwaste Tank Rupture | *** |
| 33. | Control Rod Drop Accident | *** |
| 34. | Pipe Breaks Inside the Primary Containment (Loss of Coolant Accident) | *** |
| 35. | Pipe Breaks Outside Primary Containment (Steam Line Break Accident) | * |
| 36. | Fuel Handling Accident | *** |
| 37. | Feedwater System Piping Break | *** |
| 38. | Failure of Air Ejector Lines | * |

Attachment 2

## COLD CRITICALITY TESTING

| | Chapter 15 Event | Event Category | Assumes Un-availability of Offsite AC |
|---|---|---|---|
| 1. | Generator Load Rejection | * | N/A |
| 2. | Turbine Trip | * | N/A |
| 3. | Turbine Trip with Failure of Generator Breakers to Open | * | N/A |
| 4. | MSIV Closures | * | N/A |
| 5. | Pressure Regulator Failure - Open | * | N/A |
| 6. | Pressure Regulator Failure - Closed | * | N/A |
| 7. | Feedwater Controller Failure - Maximum Demand | ** | No |
| 8. | Loss of Feedwater Heating | * | N/A |
| 9. | Shutdown Cooling (RHR) Malfunction - Decreasing Temperature | ** | No |
| 10. | Inadvertent HPCI Pump Start | * | N/A |
| 11. | Continuous Control Rod Withdrawal During Power Range Operation | * | N/A |

* Event not possible.

** Event possible but essentially no consequences.

| | | | |
|---|---|---|---|
| 12. | Continuous Rod Withdrawal During Reactor Startup | ** | No |
| 13. | Control Rod Removal Error During Refueling | ** | No |
| 14. | Fuel Assembly Insertion Error During Refueling | ** | No |
| 15. | Off-Design Operational Transients Due to Inadvertent Loading of a Fuel Assembly into an Improper Location | ** | No |
| 16. | Inadvertent Loading and Operation of a Fuel Assembly in Improper Location | ** | No |
| 17. | Inadvertent Opening of a Safety/Relief Valve | * | N/A |
| 18. | Loss of Feedwater Flow | ** | No |
| 19. | Loss of AC Power | ** | Yes |
| 20. | Recirculation Pump Trip | ** | No |
| 21. | Loss of Condenser Vacuum | * | N/A |
| 22. | Recirculation Pump Seizure | ** | No |
| 23. | Recirculation Flow Control Failure - Decreasing Flow | ** | No |
| 24. | Recirculation Flow Control Failure With Increasing Flow | ** | No |
| 25. | Abnormal Startup of Idle Recirculation Pump | ** | No |
| 26. | Core Coolant Temperature Increase | ** | No |
| 27. | Anticipated Transients Without Scram (ATWS) | ** | No |
| 28. | Cask Drop Accident | * | N/A |
| 29. | Miscellaneous Small Releases Outside Primary Containment | ** | No |

| | | | |
|---|---|---|---|
| 30. | Off Design Operational Transient as a Consequence of Instrument Line Failure | ** | No |
| 31. | Main Condenser Gas Treatment System Failure | * | N/A |
| 32. | Liquid Radwaste Tank Rupture | ** | No |
| 33. | Control Rod Drop Accident | ** | No |
| 34. | Pipe Breaks Inside the Primary Containment (Loss of Coolant Accident) | ** | Yes |
| 35. | Pipe Breaks Outside Primary Containment (Steam Line Break Accident) | * | N/A |
| 36. | Fuel Handling Accident | ** | No |
| 37. | Feedwater System Piping Break | ** | Yes |
| 38. | Failure of Air Ejector Lines | * | N/A |

5% POWER

| | Chapter 15 Event | Event Category | Assumes Un-availability of Offsite AC |
|---|---|---|---|
| 1. | Generator Load Rejection | * | N/A |
| 2. | Turbine Trip | ** | No |
| 3. | Turbine Trip with Failure of Generator Breakers to Open | * | N/A |
| 4. | MSIV Closures | ** | No |
| 5. | Pressure Regulator Failure - Open | ** | No |
| 6. | Pressure Regulator Failure - Closed | ** | No |
| 7. | Feedwater Controller Failure - Maximum Demand | ** | No |
| 8. | Loss of Feedwater Heating | ** | No |
| 9. | Shutdown Cooling (RHR) Malfunction - Decreasing Temperature | ** | No |
| 10. | Inadvertent HPCI Pump Start | ** | No |
| 11. | Continuous Control Rod Withdrawal During Power Range Operation | ** | No |

* Event cannot occur.

** Bounded by same event at higher power level per FSAR Chapter 15.

| | | | |
|---|---|---|---|
| 12. | Continuous Rod Withdrawal During Reactor Startup | ** | No |
| 13. | Control Rod Removal Error During Refueling | ** | No |
| 14. | Fuel Assembly Insertion Error During Refueling | ** | No |
| 15. | Off-Design Operational Transients Due to Inadvertent Loading of a Fuel Assembly Into an Improper Location | ** | No |
| 16. | Inadvertent Loading and Operation of a Fuel Assembly in Improper Location | ** | No |
| 17. | Inadvertent Opening of a Safety/Relief Valve | ** | No |
| 18. | Loss of Feedwater Flow | ** | No |
| 19. | Loss of AC Power | ** | Yes |
| 20. | Recirculation Pump Trip | ** | No |
| 21. | Loss of Condenser Vacuum | ** | No |
| 22. | Recirculation Pump Seizure | ** | No |
| 23. | Recirculation Flow Control Failure - Decreasing Flow | ** | No |
| 24. | Recirculation Flow Control Failure - With Increasing Flow | ** | No |
| 25. | Abnormal Startup of Idle Recirculation Pump | ** | No |
| 26. | Core Coolant Temperature Increase | ** | No |
| 27. | Anticipated Transients Without Scram (ATWS) | ** | No |
| 28. | Cask Drop Accident | * | N/A |
| 29. | Miscellaneous Small Releases Outside Primary Containment | ** | No |

| | | | |
|---|---|---|---|
| 30. | Off Design Operational Transient as a Consequence of Instrument Line Failure | ** | No |
| 31. | Main Condenser Gas Treatment System Failure | ** | No |
| 32. | Liquid Radwaste Tank Rupture | ** | No |
| 33. | Control Rod Drop Accident | ** | No |
| 34. | Pipe Breaks Inside the Primary Containment (Loss of Coolant Accident) | ** | Yes |
| 35. | Pipe Breaks Outside Primary Containment (Steam Line Break Accident) | ** | Yes |
| 36. | Fuel Handling Accident | ** | No |
| 37. | Feedwater System Piping Break | ** | Yes |
| 38. | Failure of Air Ejector Lines | ** | No |

## LOCA Analysis at 1% and 5% Core Power

| Assumptions | Peak Rod LHGR (KW/ft.) | Time to 10 CFR 50.46 limits (Min.) | 10 CFR 50.46 Limits: PCT °F (Limit 2200°) | Local Oxidation (Limit 17%) | Core Wide Oxidation (Limit 1%) |
|---|---|---|---|---|---|
| core power - 1% | | | | | |
| Standard conservative Appendix K assumptions | 0.268 | 370 | 2200 | 11.4 | 0.8 |
| Standard Appendix K models using some realistic assumptions including decay heat and natural convection | 0.18 | Maximum cladding temperature less than 1600 F at 24 hours | -- | -- | -- |
| core power - 5% | | | | | |
| Standard conservative Appendix K assumptions | | | | | |
| -- conservative core peaking factor | 1.34 | 55 | 2200 | 6.5 | <0.9 |
| -- planned core peaking factor | 0.9 | 86 | 2200 | 7.4 | <1.0 |
| -- anticipated core power history and planned core peaking factor | 0.9 | 110 | 2200 | 8.5 | <1.0 |
| Standard Appendix K models using some realistic assumptions including decay heat and natural convection | 0.9 | > 3 hours | 2200 | <17 | <1.0 |

Attachment 4