UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of)

LONG ISLAND LIGHTING COMPANY) Docket No. 50-322-OL-4 (Low Power)

(Shoreham Nuclear Power Station,)

Unit 1)

TESTIMONY OF WAYNE HODGES

Q. What is your name?

A. My name is Marvin Wayne Hodges.

Q. What is your position at the NRC?

A. I am employed as a Section Leader in Section B of the Reactor Systems Branch in the Division of Systems Integration.

Q. What are your technical qualifications?

A. I graduated from Auburn University with a Mechanical Engineering Degree in 1965. I received a Master of Science Degree in Mechanical Engineering from Auburn University in 1967. I am a registered professional engineer in the State of Maryland (No. 13446).

In my present work assignment at the NRC, I supervise the work of six graduate engineers. My section is responsible for the review of primary and safety systems for boiling water reactors. I have served as principal reviewer in the area of boiling water reactor systems. I have also participated in the review of analytical models used in the licensing evaluations of boiling water reactors and I have the technical review responsibility for many of the modifications and analyses being implemented on boiling water reactors post Three Mile Island Unit 2 accident.

As a member of the Bulletins and Orders Task Force, which was formed after the TMI-2 accident, I was responsible for the review of the capability of BWR systems to cope with loss of feedwater transients and small-break-loss of coolant accidents.

I have also served at the NRC as a reviewer in the Analysis Branch of the NRC in the area of thermal hydraulic performance of the reactor core. I served as a consultant to the RES representative to the Program Management Group for the BWR blowdown emergency core cooling program.

Prior to joining the NRC staff in March 1974, I was employed by E. I. DuPont at the Savannah River Laboratory as a research engineer. At SRL I conducted hydraulic and heat transfer testing to support operation of the reactors at the Savannah River Plant. I also performed safety limit calculations and participated in the development of analytical models for use in transient analyses at Savannah River. My tenure at SRL was from June 1967 to March 1974.

From September 1965 to June 1967, while in graduate school, I taught courses in thermodynamics, statics, mechanical engineering measurements, computer programming, and assisted in a course in the history of engineering. During the summer of 1966, I worked at the Savannah River Laboratory doing hydraulic testing.

Q. Do NRC regulations limit peak cladding temperatures in case of accidents?

A. For loss of coolant accidents, Title 10, Paragraph 50.46 of the Code of Federal Regulations gives five limits to be satisfied. First, the calculated maximum fuel element cladding temperature shall not exceed 2200°F. Second, maximum cladding oxidation shall nowhere exceed 17% of the total cladding thickness before oxidation. Third, the calculated total amount of hydrogen generated from chemical reaction of the cladding with water or steam shall not exceed 1% of the hypothetical amount that would be generated if all the metal in the cladding cylinder surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react. Fourth, calculated changes from core geometry shall be such that the core remains amenable to cooling. Fifth, after any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

Q. If all alternating current electric power were lost when the reactor was at 5% rated power, how long would it take before a maximum fuel element cladding temperature of 2200°F would be reached in the case of an accident that caused a loss of coolant and one that did not?

A. A loss of coolant accident is the most serious accident or transient that could occur without the availability of AC power because it could lead to lack of power to drive pumps necessary to maintain water in the reactor vessel to cool the core. For a non-loss-of-coolant accident, there would be a very slow boil off of the water in the vessel. The level would drop from the normal range down to the top of the fuel over an extended period of time. Starting from the normal water level, there are 158,000 pounds of water above the top of the fuel. That is equivalent to approximately 18,930 gallons. At five minutes after a reactor trip, if all the decay heat goes to boil the water, then the required makeup to replace the water boiled away would be about 42 gallons per minute. After eight hours that value has dropped to 12 gallons per minute. However, not all of the decay heat goes to boil the water away. Some is transferred to the containment and surroundings. If either the Reactor Core Isolation Cooling system or the High Pressure Coolant Injection system acts to restore the water level to the normal range, at least once during the first four days, then heat losses to the ambient through the reactor vessel wall to the containment and out through the environment will equal the decay heat being generated before the fuel would ever uncover. For that condition, the boiloff would cease, the transfer of the heat through the reactor vessel walls would tend to depressurize the reactor vessel slowly, and a peak cladding temperature of 2200°F would never be reached. In fact, the temperature of both the fuel and the cladding would remain near the saturation temperature of the water.

For the loss of coolant accident, LILCO has performed several calculations. They did a calculation using very limiting assumptions on the power and operating history. Using an approved evaluation model for a loss of coolant accident with no makeup at all, there are approximately 55 minutes before the peak cladding temperature would exceed the 2200°F limit. For a more realistic analysis which uses more realistic peaking factors and to some extent considers the limited operating lifetime at 5%, but still using the approved evaluation model, LILCO calculates that it would take 110 minutes for the peak cladding temperature to reach the 2200°F limit. Using best estimate models, which have been reviewed and approved by the NRC, it would take more than three hours for the peak cladding temperature to exceed 2200°F (at the end of three hours the temperature would still be less than 1900°F). For all three cases analyzed, no fuel failures were predicted up to the times the cladding was calculated to reach 2200°F. These calculations were done for LILCO by the General Electric Company and have been done with an NRC approved model. Although I did not review all of the details of the specific calculations, I have reviewed the evaluation models that have been used to perform the calculations and I have reviewed major assumptions used in the calculations. I am satisfied that these are bounding calculations.

Q. What would happen if the 2200°F temperature limit were exceeded at 5% rated power?

A. That depends on the extent to which that limit is exceeded. Nothing drastic happens at 2200°F. In fact there are some data that indicate that you could go as high as 2700°F, not melt the fuel, and still retain some ductility to the cladding. The 2200°F limit was chosen as a conservative value to assure that the ductility of the cladding still exists so that following reflooding of the fuel with cold water you won't shatter the fuel and you can maintain a coolable geometry. 10 C.F.R. 50.46 also has limits on the cladding oxidation and the hydrogen generation as well as temperature limit. For the type of event we're discussing, which would be a loss of coolant accident from 5% power, there would be a very slow heatup of the fuel rods. The oxidation that can occur is a function of the time at which you're at high temperatures; also the rate of the oxidation increases as the temperature increases. Therefore, if you exceeded the 2200°F limit, you might also exceed the oxidation limit and cladding brittlement would become a concern.

Q. For the loss of coolant accident at 5% power, what is the rod internal pressure prior to reaching 2200°F and what is the significance of that value?

A. LILCO calculated a rod internal pressure of 97.7 psia at 2200°F. This is the highest value of internal pressure reached during the 55 minute heatup. For an internal pressure of 98 psia, a temperature of 2300°F would be needed to cause the cladding to rupture. At 2200°F, the rupture pressure is 117 psia. Therefore, even using the very conservative bounding analysis, no fuel rod rupture is expected. This means that there should no large release of activity because the cladding retains the fission products.

Q. What local oxidation resulted from the rod heatup in that analysis and what is its significance?

A. The maximum local oxidation was calculated to be 6½%. Using the Baker-Just equation, for values less than 17%, the cladding remains ductile and should not fracture due to thermal stresses when the fuel is quenched by cold water. Therefore, the core remains coolable. Because there is no cladding rupture, the fission products are retained in the fuel.

Q. If Shoreham were operating at 5% of rated power with qualified TDI diesels and there was a LOCA, what would the peak cladding temperature and oxidation be?

A. The peak cladding temperature has been calculated by GE to be 550°F. The local oxidation would be .033% and the core wide oxidation would be .033%.

Q. How does this compare with the LOCA with no qualified diesels and loss of normal offsite power?

A. If it is assumed that the 20 MW gas turbine fails and the GM EMDs are started in 30 minutes, the peak cladding temperature is calculated to be 1086°F, local oxidation would be .05% and core wide oxidation would be .034%. Even using a very conservative peaking factor, there are at least 55 minutes available to restore offsite power. If AC power is restored within 55 minutes for the case with no qualified diesels, then it is as safe as the case with qualified diesels because the cladding integrity is maintained and all fission products are retained in the fuel.

Q. Is there NRC staff guidance setting out the transients and accidents to be analyzed in an FSAR Chapter 15?

A. Yes. Reg. Guide 1.70 on the standard format and content for FSARs lists the transients and accidents to be analyzed. The Standard Review Plan (SRP, NUREG-0800) provides detail on how the Staff reviews the accidents and transients listed in Reg. Guide 1.70. LILCO used the transients and accidents listed in Reg. Guide 1.70 in its analysis of possible low-power transients and accidents in its submittal.

Q. We have previously talked of an accident involving the loss of all electric power with the Shoreham reactor operating at 5% of rated power. How does this compare with the spectrum of transients and accidents set out in Chapter 15 of the FSAR?

A. Except for the loss of coolant accident, all of the transients and accidents analyzed in the FSAR, even with no alternating current power available for the 5% power case, are less restrictive than for the design basis cases analyzed in Chapter 15 of the FSAR. The loss of coolant analysis has been discussed previously. The review of the FSAR Chapter 15 analysis shows that of the 38 accident or transient events addressed in Chapter 15, 5 events cannot occur during this phase. Generator load rejection and turbine trip with failure of generator breakers to open events are not possible because the generator will not be connected to the grid. Control rod removal errors during refueling are precluded by definition. A cask drop accident is precluded by design, hence it is not postulated in the analysis. The remaining 33 events are considered.

For all of the events, operation of the plant up to 5% rated power will be bounded by the Chapter 15 analysis. For example, the turbine trip event is analyzed with the assumption that the limiting event occurs with the reactor operating at 105% of rated steam flow coupled with failure of the turbine bypass valves to open. Even this limiting event does not result in any fuel failures. The FSAR specifically notes that turbine trips at power levels less than 30% of rated power are bounded by the limiting analysis. Another example is the loss of feedwater heating event. This event is analyzed with the assumption of continuous operation of the feedwater system and the most severe possible loss of feedwater heating, resulting in the injection of colder feedwater. For operation at power levels less than 5% , the impact of lost feedwater heating is minimal because of the low feedwater flow.

For low power testing up to 5% power, the fission product inventory in the core will not exceed 5% of the values assumed in the FSAR. In addition, because of the small temperature differential across the pellets, almost all of the fission products will be retained in the pellets. LILCO estimates that the fuel burnup during low power testing will be less than 200 MWD/MTU (REF: LILCO letter SNRC-1036 dated April 11, 1984). This low fuel burnup enhances safety in three ways: (a) the amount of decay heat present in the core following shutdown is substantially reduced resulting in reduced cooling system requirements, (b) the amount of radioactivity that could be released upon fuel failure is substantially (more than a factor of 20) reduced, (c) and if additional failures were postulated to occur, the operator will have longer time to take corrective actions.

Another factor contributing to enhanced safety during low power operation is the increased time available for preventive or mitigating action should such action be deemed desirable by the operator. Longer time is available because the limited power levels mean that it takes longer for the plant to reach setpoints and limits. For example, on loss of feedwater, the water level in the reactor will decrease at a slower rate than if the event occurred at 100% power. If HPCI or RCIC operate at least once during the first four days to restore normal water level, then no additional make up will be required to prevent core uncovery due to boil-off. Similarly, in the loss of condenser vacuum event, the operator will have more time to identify the decreasing vacuum and to take steps to remedy the situation before automatic actions such as turbine trip, feed pump trip or main steam isolation occur. Another example is the main steam isolation valve closure event. At five percent power, the amount of heat produced upon isolation of the reactor vessel (which is followed by a reactor trip) 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 cooling rather than relying on automatic action. In effect, the operator may end the transient before there is any substantial impact on the plant.

Another factor contributing to the enhanced safety during lower power testing is the reduction in the required capacity for mitigating systems. Because of the 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. (Five minutes after shutdown, about 42 GPM makeup is required to compensate for boil-off; after 8 hours, 12 GPM are required).

Q. If fuel were loaded in the reactor, the reactor had not reached criticality, and all electric power were lost, how long a period of time would there be before any fuel rod reached 2200°F?

A. Because there would be no nuclear heat generation, there would be no heatup of the fuel so that even if all the water were lost from the vessel and there were no water makeup, the fuel would sit in the vessel and the temperature would remain near ambient. You would not reach 2200°F.

Q. Is the availability of AC power a concern if criticality had not been reached?

A. Availability of AC power is not a safety concern because many of the transients cannot occur and for those that can occur, there can be no radiological consequences regardless of whether or not AC power is available. Therefore, there is no risk to the public health and safety.

Q. If the reactor were operating at .001% power as described in Phase II of the LILCO low power submission, and all alternating current power were lost, could a LOCA occur?

A. For conditions described in Phase II where the reactor is operating at essentially ambient temperature and pressure, there are not stresses in the piping system great enough that a loss of coolant accident would be concern, so it is extremely unlikely that a LOCA would occur under these conditions. However, if a loss of coolant accident should occur during Phase II testing, LILCO has calculated that there would be time on the order of months available to restore makeup water for core cooling. At the decay heat levels which would exist under these conditions, heat transfer to the environment would remove a significant fraction of the decay heat and it is likely that the fuel would never heat up to 2200°F. However, even if no heat transfer from the fuel rod is assumed, so that you have an adiabatic heatup of the fuel rod, and equilibrium fission products are assumed for infinite operation at .001% power, I calculate that more than 30 days are available to restore cooling prior to exceeding a fuel rod temperature of 2200°F.

Q. How do the accidents and transients possible during Phase II compare to those set out in Chapter 15 of the Shoreham FSAR?

A. The review of anticipated operational occurrences and postulated accidents set out in Chapter 15 of the Shoreham FSAR, when compared to Chapter 15 Phase II operation indicates that most of the transients are not possible for the same reasons described in the Phase I evaluation. Because the fission products inventories in the core will be significantly less during Phase II operation than for conditions analyzed in the FSAR, the radiological impact for transients involving continuous control rod withdrawal during startup event, fuel handling accident, liquid radwaste tank rupture are significantly less severe than those that have already been analyzed and found acceptable in the FSAR.

Q. Is the availability of the AC power a concern during LILCO's projected Phase II operation?

A. Availability of AC power is not a safety concern during Phase II, because many of the transients cannot occur and for those that can occur, it very unlikely that fuel failure could occur. Even if it did, there can be no significant radiological consequences due to very low fission product inventory. Therefore, there is no significant risk to the public health and safety.

Q. What plant systems need power to keep the hottest fuel rod from going over 2200°F in the event of an accident?

A. For transients which do not depressurize the vessel, either the reactor core isolation cooling system, the high pressure coolant injection system, both of which are steam driven, or the control rod drive system would be sufficient to maintain water inventory. The fuel would remain covered with water and would not heat up. For a design basis accident where all of the water inventory would be initially removed from the vessel and there would be no steam available to drive the RCIC or the HPCI and supply water to cool the core, you would need a core spray system or a low pressure coolant injection system to provide water to flood the core up to the 2/3 core height. However, even for the LOCA case there are on the order of 55 minutes available before the maximum fuel element cladding temperature exceeded 2200°F and power had to be restored.

Q. In your answers, do you assume that a LOCA and a seismic event occur simultaneously?

A. Although the equipment which is used to mitigate a loss of coolant accident is normally required to satisfy seismic criteria, the Staff does not assume the simultaneous occurrence of a loss of coolant accident and a seismic event. This is because of the very low probability of the combined event.

Q. We have previously talked about fuel rod temperatures. Is this the bounding source of any of the concerns for an accident starting from 5% at Shoreham?

A. For the cases that we have discussed, the peak cladding temperature would be reached prior to any of the other limits that are described in 10 C.F.R. 50.46. For a lower power condition such as operation at 1 or 2%, it is possible that an oxidation limit could be reached prior to reaching the fuel temperature limit. However, in either case, the 55 minutes that's been described for the 5% case would bound the time available to restore power to prevent reaching any of these limits.