

1 UNITED STATES OF AMERICA
2 NUCLEAR REGULATORY COMMISSION

3 BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

4
5 In the Matter of) Docket Nos. 50-361 OL
6 SOUTHERN CALIFORNIA) 50-362 OL
7 EDISON COMPANY, ET. AL.)
8 (San Onofre Nuclear Generating)
9 Station, Units 2 & 3))
10
11
12
13
14
15

16 DIRECT TESTIMONY
17 OF MR. DAVID R. BUTTNER
18
19
20
21
22
23
24
25
26

TESTIMONY OF DAVID R. BUTTEMER

Q. Would you please state your name?

A. David R. Buttemer.

Q. By whom are you presently employed?

A. I am employed as Senior Consultant by Pickard, Lowe and Garrick, Inc. (PLG), a consulting engineering firm specializing in reactor siting, safety, and accident analysis.

Q. For what purposes has PLG been retained by the Applicants in this proceeding?

A. PLG was retained by the Applicants to assess the response of the San Onofre Nuclear Generating Station Unit 2 (SONGS 2) to a range of postulated accidents which might occur during low power testing.

Q. In what manner have you been involved in the work conducted by PLG for the Applicants?

A. I have been involved in performing a variety of safety analyses of potential low power accidents at the SONGS 2 plant. During the course of these studies I have been in direct contact with the engineering staff of the nuclear steam supply vendor for this plant, Combustion Engineering, Inc., and obtained detailed information regarding the reactor coolant system. The results of my analyses are set forth in

////

////

1 Exhibit DRB-1, "Analysis of Postulated Accidents During Low
2 Power Testing at the San Onofre Nuclear Generating Station,
3 Unit 2".

4 Q. What are your pertinent professional affiliations?

5 A. I am a registered Professional Engineer in the State of
6 California with certificates in both mechanical and nuclear
7 engineering.

8 Q. Please describe your academic qualifications pertinent to
9 reactor accident analysis?

10 A. I graduated in 1960 from San Diego State University with a
11 Bachelor of Science Degree in Mechanical Engineering. I
12 received a Master of Science Degree in Mechanical Engineering
13 from UCLA in 1965. In the summer of 1976 and 1977, I was a
14 lecturer at the Fast Reactor Safety Course given at the
15 Massachusetts Institute of Technology.

16 Q. What professional experience have you had in the field of
17 safety and accident analysis?

18 A. During the period 1960 through 1977, I was employed by the
19 General Atomic Company. From 1960 to 1965, I was principally
20 involved in the mechanical design of the reactor core of an
21 experimental reactor. From 1966 through 1968, I was
22 principally involved in the analysis and design of major High
23 Temperature Gas-Cooled Reactor components including the
24 reactor core, steam generator and pre-stressed concrete
25 reactor vessel. From 1969 to 1977, I was responsible for the
26 safety and systems analysis of the gas-cooled fast breeder

1 reactor (GCFR) in the capacity of Branch Manager. This work
2 entailed a wide range of accident analyses, including
3 analytical methods development and analysis of accidents well
4 beyond the so-called design basis. In 1973, I spent six
5 months in Germany as a consultant to a German reactor
6 manufacturer, Kraftwerk Union, and to the German national
7 laboratory at Karlsruhe training their staffs on the use of
8 large accident analysis computer programs I had developed
9 while at General Atomic. From 1977 to 1980, I was employed
10 by Helium Breeder Associates ("HBA"), a firm responsible for
11 providing utility industry management and financial support
12 to the GCFR program. While at HBA, I was the Technical
13 Director and was also the Manager of the Technical Division.
14 In this capacity I was responsible for the overall technical
15 direction of the program, working closely with the U.S.
16 Department of Energy in establishing priorities and
17 coordinating work being done by General Atomic, several U.S.
18 national laboratories and several architect/engineering
19 firms. The U.S. Department of Energy discontinued financial
20 support of the GCFR program in 1980 and I joined the
21 consulting firm of Pickard, Lowe and Garrick, Inc., ("PLG")
22 in December of 1980. At PLG, I have been primarily involved
23 in the area of probabilistic risk assessment.

24 Q. Have you previously submitted expert opinions in the area of
25 safety analysis?

26 ////

- 1 A. Yes. In the time period of 1971 to 1975, I made numerous
2 presentations on GCFR safety and licensing analyses to the
3 Nuclear Regulatory Commission Staff and the Advisory
4 Committee on Reactor Safety.
- 5 Q. What is the purpose of your testimony in this proceeding?
- 6 A. My testimony is in support of Applicants' Alternative Motion
7 For a Fuel Loading and Low Power License For SONGS 2. My
8 testimony demonstrates that even extremely unlikely accidents
9 well beyond the design basis for SONGS 2, would progress very
10 slowly under the low power test program restraints. In
11 conjunction with the testimony of Rosenblum and Pilmer, my
12 testimony will show that such accident sequences progress
13 sufficiently, slowly to allow taking effective action to
14 prevent serious accidents or to take offsite protective
15 actions.
- 16 Q. Would you describe how you have used the concept of "risk" in
17 performing your evaluation?
- 18 A. Yes. My studies were made to evaluate the risks associated
19 with fuel loading and low power operation relative to those
20 associated with full power operation. In this context risk
21 is comprised of two principal components. First, the
22 probability, or likelihood, that a given accident sequences
23 will occur, and, secondly, the public consequences associated
24 with that sequence.
- 25 Q. Please describe the factors affecting potential public
26 consequences which you consider could conceivably occur

1 during low power testing as compared to full power
2 operation?

3 A. A major factor affecting public consequences is the
4 inventory, or amount, of radioactive nuclides available at
5 the plant. In a reactor facility which has been operating at
6 full power for an extended period of time, by far the largest
7 radionuclide inventory is in the reactor core itself,
8 although significant inventories also exist in the spent fuel
9 located in the spent fuel storage pool as well as in the
10 radioactive waste systems and activated corrosion products.
11 During the planned SONGS 2 low power test program there will
12 be no spent fuel, very little radioactivity in the
13 radioactive waste systems and essentially no activated
14 corrosion products.

15 Because of the low reactor power levels and short
16 operating times planned in the low power test program, the
17 fission product inventory within the core itself is a small
18 fraction of that which would exist during normal operation.
19 Short-lived fission product inventories would be about 1/20th
20 of that which would occur during normal operation. The
21 longer-lived fission product inventories would be less than
22 1/20th of that during normal power operation.

23 The substantially lower core fission product
24 inventories represent a much lower radionuclide source term
25 in the context of accident dose, more importantly,
26 represent substantially lower fission product decay heat

1 levels. In the event of an accident, the lower decay heat
2 results in very slow heat up rates providing substantial time
3 for mitigative action. At 5% power the core temperatures are
4 much lower than at full power, the stored thermal energy in
5 the core being about 5% of that at full power. These factors
6 provide much greater thermal margins to the design limits,
7 which are established based upon full power operation.

8 Q. Have you identified any factors peculiar to low power
9 testing which would increase the potential accident
10 consequences relative to full power operation?

11 A. No.

12 Q. How do the probabilities of severe accident sequences during
13 low power testing compare to those at full power?

14 A. The probability that an accident will be initiated during
15 low power operation should be about the same as during full
16 power operation. Bear in mind that probability expresses the
17 likelihood that a given event will occur during a given
18 period of time. The SONGS 2 plant will be at power greater
19 than 0.1 percent for only about two weeks during the low
20 power test phase. From the Reactor Safety Study, The Wash
21 1400 Report issued by the AEC in 1975, accident initiators
22 are grouped into two broad categories - loss of coolant
23 accidents ("LOCAs") and transients. Since the low power
24 tests are conducted at full pressure and at coolant
25 temperatures comparable to those at full power, the
26 probability of a LOCA would be about the same during low

1 power testing as at full power. Transient accident
2 initiators are caused by a wide range of events, many of
3 which are associated with operator error or failures in the
4 turbine generator portion of the plant. The low power test
5 will be conducted under strict procedural controls under the
6 direct scrutiny of engineering and technical supervisors.
7 Also, during low power physics and natural circulation tests,
8 the turbine generator system will not be operating, feedwater
9 will be supplied to the steam generators by the safety grade
10 auxiliary feedwater system and the steam produced in the
11 steam generators will be condensed in the main condenser.
12 For these reasons, the likelihood of a transient accident
13 initiator would be lower than during full power operation.

14 Q. Have you identified any factors which would increase the
15 likelihood of accident initiators?

16 A. Yes. Because of the newness of the plant, somewhat higher
17 equipment break in failures are possible, and some
18 uncertainties in integrated system performance exist.
19 However, the plant systems have been rigorously tested over a
20 period of several years as part of the system and hot
21 functional test programs. Although the maintenance,
22 operating, and emergency procedures have been utilized in the
23 startup program, some further refinement may be required.

24 Q. What is the net effect of the above factors on the likelihood
25 of transient accident initiators?

26 ////

1 A. All in all, I would expect that the likelihood of transient
2 accident initiators would be about the same for either low
3 power or full power operation.

4 Q. You have discussed the relative probabilities of initiation
5 of an accident. Assuming an initiator has occurred, would
6 the accident progress differently at low power than at full
7 power?

8 A. Yes. All of the safety systems which are designed to
9 accommodate accidents from full power operation would be
10 available for action, if needed, during the low power testing
11 phase. These safety grade systems would be expected to
12 automatically start and avert excessive core temperatures
13 should an accident occur. If for some unforeseen reason
14 these systems should malfunction, excessive core temperatures
15 will not be reached for several tens of hours, thereby
16 allowing ample time for diagnostic and corrective operator
17 action.

18 Q. What is the net effect of these factors on the likelihood of
19 severe accident sequences?

20 A. For the above reasons, I expect that during low power
21 testing, the probability of accident sequences leading to
22 core melt would be lower than at full power operation.

23 Q. Having examined both the probability and consequences of
24 severe accidents during low power testing relative to full
25 power operation, what is your assessment of the relative risk
26 of low power testing?

1 A. Because both the probability and consequences of core melt
2 accidents are smaller during low power testing than at full
3 power, I conclude that the risk is also much lower.

4 Q. Your Exhibit DRB-1 "Analysis of Postulated Accidents During
5 Low Power Testing at San Onofre Nuclear Generating Station,
6 Unit 2" presents a detailed analysis of various accident
7 scenarios. Can you summarize the significant conclusions of
8 that evaluation?

9 A. Yes. At the time these analyses were begun, the detailed low
10 power testing program was not available. Therefore, I
11 assumed continuous operation at 5% power for time periods of
12 two and a half, five and ten days. I considered three
13 classes of accident initiators: (1) a large LOCA, (2) an
14 instantaneous loss of the steam generator as a heat sink with
15 the safety valves remaining open, and (3) an instantaneous
16 loss of the steam generator as a heat sink with the safety
17 valves maintaining reactor coolant pressure at the 2500 psia
18 set point. In the loss of steam generator heat sink
19 accidents, no credit was taken for boiloff of the secondary
20 water in the steam generators themselves. The reactor was
21 assumed to be shut down after the accident but it was assumed
22 that the active safety systems, mainly the auxiliary
23 feedwater and emergency core cooling systems, were not
24 operational. This is an extremely conservative assumption,
25 in that these analyses apply to accident sequences which have
26 an exceedingly small probability of

1 occurrence. The core decay heat and integral decay heat were
2 evaluated as functions of time after scram for the three
3 operating periods assumed. Next, a thermal hydraulic model
4 of the core in the steam cooling phase was developed. This
5 is the phase when the water level in the reactor vessel is in
6 the active core region. For the large LOCA analysis it is
7 important to determine how much of the water initially in the
8 reactor coolant system and in the safety injection tanks
9 ("SIT") is available for boiloff. The water level cannot be
10 above the reactor vessel nozzles since it would spill out the
11 severed pipe. For certain classes of LOCA's, it is possible
12 that residual nitrogen gas pressure in the safety injection
13 tanks can displace water out of the vessel leak as it is
14 vented. This factor was taken into consideration. In the
15 loss of steam generator cases, steam pockets can form in the
16 upper extremities of the reactor coolant system and displace
17 water out of the safety valves located on the pressurizer.
18 This factor was also accounted for. The residual water will
19 then be heated up and boiled off by the heat generated in the
20 core as well as by any stored heat remaining in the reactor
21 vessel and its internals after the blowdown-SIT injection
22 phase has ended. The core temperatures will gradually
23 increase as the water level recedes and the time when
24 excessive core temperatures are reached is of interest. This
25 temperature has been conservatively selected as 1,560 degrees
26 Fahrenheit, the temperature at which significant metal water

1 reaction would begin. These times are indicated in Figure
2 DRB-A, "Summary of Accident Analyses--SONGS Unit 2, Low Power
3 Testing Program" for the three accident initiators considered
4 as a function of the days of prior continuous operation at 5%
5 power. As can be seen, these times are very long. The large
6 LOCA is the most limiting accident. With 10 days prior
7 operation it can be seen that excessive core temperatures are
8 not reached for 22 hours. For the loss of steam generator
9 heat sink events, excessive temperatures are not reached for
10 several days. Also shown in Figure DRB-A are the water make
11 up rates which would avert excessive core temperatures.
12 These make up rates are very small. The very long times
13 before the onset of core damage and the small water makeup
14 requirements allow for adequate corrective action to be taken
15 to arrest the accident sequence.

16 Q. Your analysis assumed no corrective action is taken up to the
17 point that excessive core temperatures are reached. What
18 happens if no corrective actions are taken subsequent to
19 reaching such excessive core temperatures?

20 A. As the temperature in the core heats up past 1560 degrees
21 Farenheit, an oxidation process begins to occur between the
22 hot cladding and the steam. This chemical reaction, referred
23 to as a metal-water reaction, produces hydrogen as one of the
24 reaction products. Since hydrogen is a combustible gas, one
25 is concerned with the rate at which it accumulates within the
26 containment building. A hydrogen/air mixture will burn

1 extensively (but will not explode) when the hydrogen reaches
2 about 8 percent by volume. The hydrogen generation rate was
3 conservatively evaluated for the large LOCA case assuming 10
4 days prior operation. This analysis shows that an additional
5 17 hours of oxidation is required to produce 8 volume percent
6 hydrogen within the containment. Therefore, this will occur
7 a total of about 40 hours after the accident is initiated,
8 assuming no containment sprays or hydrogen recombiners
9 operate. Additionally, the resulting calculated peak
10 pressure after hydrogen burn is less than the pressure at
11 which containment structural integrity is jeopardized and
12 therefore no release of radioactive material would be
13 contemplated.

Reactor Power History

- o 2.5 days continuous operation at 5% power
- o 5 days continuous operation at 5% power
- o 10 days continuous operation at 5% power

Postulated Accidents (all active systems assumed to fail)

- o Large Loss-of-Coolant-Accident (LOCA)
- o Loss of Steam Generator Heat Sink Accident - Safety valves remain open
- o Loss of Steam Generator Heat Sink Accident - Safety valves maintain pressure at 2500 psia

Times for Excessive Core Temperatures and Water Makeup

Accident Initiator	Days Prior Operation	Time When Significant Clad Metal-Water Reaction Begins (TCLAD > 1,560°F)	Water Makeup Required To Prevent Excessive Core Temperatures
LOCA	2.5 5.0 10.0	47 hrs 28 hrs 22 hrs	0.8 gpm 1.5 gpm 2.2 gpm
LOSGHSA* - SVs remain open	2.5 5.0 10.0	11.0 days 5.6 days 3.7 days	0.2 gpm 0.6 gpm 1.0 gpm
LOSGHSA - SVs Maintain Pressure at 2,500 psia	2.5 5.0 10.0	12.2 days 5.8 days 3.9 days	0.4 gpm 0.8 gpm 1.5 gpm

*LOSGHSA is loss of steam generator heat sink accident. SVs are safety valves.

Figure DRB-A: "Summary of Accident Analyses - SONGS Unit 2 Low Power Testing Program"