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May 26, 1982

Director, Office of Nuclear Reactor Regulation
Attention: D. M. Crutchfield, Chief
Operating Reactors Branch No. 5
Division of Licensing
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Gentlemen:

Subject: Docket No. 50-206
Pressurized Thermal Shock to Reactor Pressure Vessels
San Onofre Nuclear Generating Station
Unit 1

By letter dated March 15, 1982, you requested that we provide you with additional information regarding the subject issue. In particular, your request related to information provided in our 150 day response transmitted to you by letter dated January 26, 1982 and a review of our plant operating history. The information you have requested is partially provided in the enclosure to this letter. The responses which have not been provided are to be included in the Westinghouse Owner's Group submittals scheduled to be submitted to the NRC on May 28, 1982 and by mid-June, 1982. These items are clearly identified in the enclosure.

A review of our plant operating history has identified three potential PTS events which have occurred. Details concerning these events are discussed in the enclosure. The three events were not significant enough to pose a PTS concern to the reactor vessel.

If you have any further questions, please let me know.

Very truly yours,

K P Baskin

Enclosure

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Enclosure

Southern California Edison's Response To

NRC Request For Additional Information

Concerning

Pressurized Thermal Shock

And

Regarding the "150" Day Response To NRC Letter Dated August 21, 1981

For

San Onofre Unit 1

Docket No. 50-206

1. Provide the following information related to fluence determination:

   (A) Plant specific information which would allow determination of the pressure vessel fluence. Such information should contain as built core and pressure vessel dimensions, regional material composition and neutron source for a two-dimensional (R-O) and (R-Z) neutron transport solution, and

   (B) Plant specific values of the pressure vessel fluence and its estimated uncertainty.

Response:

(A) As built core and pressure vessel dimensions are not available. Nominal design dimensions were provided in Figure 1-1 of the 150 day response submitted January 25, 1982. The use of design vs as built dimensions is not considered significant to fluence prediction based on the good agreement of calculations with surveillance capsule measurements discussed in (B) below.

Design basis reactor core power distributions which were used by Westinghouse in performing fast neutron fluence evaluations for the San Onofre Unit 1 pressure vessel were supplied to the NRC in our 150 day response. That submittal consisted of the following items:

1. The material description of the reactor core presented in terms of volume fractions of solid material within a homogenized fuel zone.

2. Individual fuel assembly power density levels within one core octant. This data was presented relative to a core average of 1.0.

3. Spatial gradients of power density within each of the peripheral fuel assemblies. This data was presented relative to an assembly average of 1.0.

4. A time averaged relative axial distribution of power density applicable to long term fast neutron fluence projections.

The methodology used to define these long term design basis power distributions was outlined in WCAP 10019, Summary Report on Reactor Vessel Integrity for Westinghouse Operating Plants, dated December, 1981. Also described in WCAP 10019 was the methodology by which these power distributions were used in subsequent neutron transport calculations.

Following submittal of WCAP 10019 and the 150 day responses, the NRC requested additional plant specific information which would "allow determination of the current pressure vessel fluence". It is our understanding that this plant specific data is not intended to be used to perform fast neutron fluence projections for future operation. Therefore, the following information is provided for use by the NRC in their evaluation of the present condition of the pressure vessel.

The material composition submitted in the 150 day response was based on a fuel assembly design consisting of a 17 x 17 array of zirconium clad fuel rods. In actuality, the San Onofre Unit 1 reactor employs a fuel design consisting of a 15 x 15 array of stainless steel clad fuel rods. A comparison of the material volume fractions for a homogenized reactor core employing each of these fuel designs is given in Table 1. An examination of Table 1 shows that the compositions of the two fuel assemblies are quite similar and in our opinion the differences will have an insignificant impact on reactor vessel fluence calculations.

Plant specific peripheral assembly power distributions for cycles 1 through 8 are tabulated in Table 2. These data were extracted from the appropriate core design reports (WCAP's 3269-07, 7490, 7799, 8060, 8490, 8933, 9334, 9633). Bias factors were applied to the design power distributions consistent with the methodology outlined in WCAP 10019. Also presented in Table 2 are the eight cycle time average power distributions for the peripheral assemblies. These average distributions were obtained by burnup weighting of the individual fuel cycle data sets. A comparison of the cycle average data with the design basis peripheral power distribution is depicted in Figure 1. An examination of Figure 1 shows that the plant specfic power distribution will result in a somewhat lower fluence projection than that which would be calculated using the design basis distribution. It would appear that a reduction in pressure vessel fluence on the order of 20% might be realized. However, at this time neutron transport calculations using the plant specific power distributions have not been carried out. These computations must be complete before any reduction in the current pressure vessel fluence can be certified. It must also be reemphasized that the plant specific data are applicable only for establishing the present condition of the pressure vessel. They should not be used to project forward in time.

An examination was also made of the variations in the power density gradients for the peripheral fuel assemblies at beginning of life and end of life for both 15 x 15 and 17 x 17 fuel rod arrays. The conclusion of this study was that these spatial gradients, relative to an assembly average power of 1.0, were quite similar in all cases examined.

Therefore, the gradient information previously provided in the 150 day response should also be used to generate plant specific fluence values for San Onofre Unit 1. Likewise, the time averaged axial power distribution supplied in the 150 day response is suitable for the current analysis.

(B) A summary of the results of the latest design basis neutron transport calculation for the San Onofre Unit 1 pressure vessel were provided in Figures 1-6 through 1-8 of the 150 day response. The estimated uncertainty in the prediction of pressure vessel fluence was discussed in WCAP-10019. It was noted that the best estimate computation with an uncertainty level of $\pm$ 20 percent bounded measured data from a large number of reactor vessel surveillance capsules. Agreement between

calculations and measurement for San Onofre Unit 1 is indicated in Table 1-3 through 1-5 of the 150 day response and is consistent with the above estimated uncertainty. A low leakage core loading pattern has been used since the removal of the last surveillance capsule in 1978. Current fluence predictions do not take credit for any reductions in fluence due to this fuel management scheme and hence are conservative.

TABLE 1

MATERIAL COMPOSITION OF REACTOR CORE REGION

MATERIAL	VOLUME FRACTION DESIGN BASIS	VOLUME FRACTION SAN ONOFRE
Water	.58864	.596
UO_2	.29967	.317
Zirc - 4	.10035	
Inconel - 718	.00281	
Stainless Steel - 304	.00062	.087

TABLE 2

SAN ONOFRE UNIT 1 PERIPHERAL POWER

ASSEMBLY No.	CYCLE 1	2	3	4	5	6	7	8	AVG.
2	.59	.77	.76	.52	.77	.77	.68	.68	.69
3	.49	.64	.63	.59	.65	.67	.48	.48	.57
4	.96	1.10	1.12	1.14	1.16	1.16	1.11	1.12	1.10
5	.76	.97	.89	.93	.94	.98	.96	.98	.92
6	.52	.71	.63	.65	.66	.71	.69	.71	.65
7	.94	1.16	1.13	1.12	1.10	1.13	1.15	1.17	1.11
8	.59	.79	.76	.68	.73	.77	.78	.79	.73
9	.91	1.06	1.13	.89	1.02	1.04	1.13	1.14	1.04
Burnup	14300	8000	10000	9650	9630	9400	10950	9950	

NOTE: THE FUEL ASSEMBLY NUMBERS REFER TO CORE POSITIONS DESIGNATED IN FIGURE 1-4 OF THE 150 DAY RESPONSE

FIGURE 1

COMPARISON OF SAN ONOFRE UNIT 1
AND DESIGN BASIS PERIPHERAL POWER DISTRIBUTIONS

0.69 0.93	0.57 0.77	SAN ONOFRE DESIGN BASIS			
	1.10 1.07	0.92 1.12	0.65 0.80		
			1.11 1.04	0.73 0.85	
				1.04 0.92	

2. Concerning Operator Action

In your evaluation the actions described do not provide the operator with clear direction for dealing with conflicting concerns that need to be evaluated when considering the operation of HPI and the charging flow as it relates to vessel integrity and maintaining core cooling. Provide an evaluation of the need and effectiveness of procedure modifications to clearly identify the concerns in the emergency operating procedures themselves. This should be done in contrast of depending upon upgrading operator training alone.

Response:

The Westinghouse Owners Group has recently completed a review of all Emergency Operating Guidelines to confirm that the guidance is consistent with current reactor vessel integrity analyses. This review has confirmed the acceptability of the existing guidelines and has identified some areas where additional clarification would be beneficial. These areas will be identified and discussed in the Westinghouse Owners Group submittal scheduled to be transmitted to the NRC in mid-June. In addition SCE has conducted an emergency operating procedure review and upgrade program in response to NRC staff concerns on the main steamline break SEP Topic. This program includes a review of the operator guidance relative to PTS concerns for loss of secondary coolant events and has resulted in some modifications to the emergency procedures. The results of the program were transmitted to the NRC by letter dated May 20, 1982.

3. Concerning Input Data and Assumptions

3.0 Provide a description of the models or data used for:

(a) Heat sources (or sinks),
(b) Decay heat,
(c) ECC and feedwater temperatures (enthalpies) and flow rates,
(d) Primary and secondary relief capacities,
(e) Empirical correlation coefficients used for PTS evaluations,
(f) Initial conditions.

Response:

The models used in the thermal-hydraulic transient analysis for San Onofre Unit 1 are the same models as those described in WCAP-10019. As indicated in our 150 day response, the models used for the small LOCA and small steamline break were generic. A description of the Westinghouse NOTRUMP code used to model the thermal hydraulic system behavior associated with the small break LOCA transient in WCAP-10019 was transmitted by Westinghouse to the NRC by letter NS-EPR-2580 dated March 30, 1982. For the large LOCA plant specific maximum SI flow rate and minimum RWST temperature were assumed (0 psi across the vessel wall is assumed at time 0). For the large steamline break, a quasi-plant specific model was used. This model used existing pressure and temperature transient profiles from other plant analyses which were judged to best represent the San Onofre Unit 1 response. Additional plant specific RV integrity evaluations were performed for the large steamline break as part of an emergency operating procedure review and upgrade program which was transmitted to the NRC by letter dated May 20, 1982.

3.2 Provide a list of all transients or accidents by class (for example: excessive feedwater, operating transients which result from multiple failures including control system failures and/or operator error, steamline break and small break LOCA) which could lead to inside vessel fluid temperatures of 300^{0}F or lower. Provide any Failure Modes and Effects Analyses (FMEAs) of control systems currently available or reference any such analyses already submitted. Estimate the frequency of occurrence of these events and provide the basis for the estimates. Discuss the assumptions made regarding reactor operator actions.

For a given initiating event, potential multiple and consequences failures need to be considered to identify those transients which could lead to a PTS problem.

Response:

As indicated in the Westinghouse Owners Group letter OG-68 dated March 23, 1982, this item will be addressed in the Westinghouse Owners Group May 28, 1982 submittal to the NRC on RV integrity. This section of the submittal will define potentially significant non-design basis transients based on frequency of occurrence or fracture mechanics considerations, provide a description of the event trees and transients considered, and provide a description of the fracture analysis performed.

3.3 Identify all potential PTS events which have occurred at your facility. Include a designation of the operator actions and identify potential additional failures (including operator) which could have resulted in a more severe event.

4. Concerning a Review of Operating History

Review your operating history at your plant and identify events which have resulted in exceeding the cooldown rate of 100^{0}F/hr. as well as those events which could have exceeded the cooldown rate limit if not mitigated by plant controls or operator actions.

Response:

In response to both Items 3.3 and 4 above the operating history at San Onofre Unit 1 was reviewed to identify any major overcooling events which have occurred. Three events were identified and are described below. All three events were a result of excess feedwater flow to the steam generators causing overfill and cooldown of the primary system. All three events were terminated when the Safety Injection setpoint was reached. The Feedwater System is isolated at this point to allow use of the feedwater pump in the Safety Injection System. Termination of the cooldown prevented RCS pressure from falling below the shutoff head of the Safety Injection System so that no actual Safety Injection flow occurred. In all three events, the total cooldown was limited to less than 100^{0}F by the termination of feedwater upon Safety Injection. Operator action to terminate the cooldown for these events

was not required due to the automatic isolation of the main feedwater system on the SI signal. If the main feedwater system fails to isolate, the emergency operating procedure requires the operator to take immediate action to verify the SI valves alignment and manually open or close the isolation valves as appropriate.

April 30, 1972

On April 30, 1972, with the unit at 55 MWe during startup from an outage, a failure of the "C" feedwater controller resulted in a reactor trip on high "C" steam generator level. Overfilling this generator caused average RCS temperature to fall from 550°F to about 460°F in 18 minutes and pressure fell from 2035 to 1550 psig. The event was terminated when the safety injection setpoint of 1685 psig was reached and safety injection initiated. No actual flow from the safety injection system was added to the reactor coolant system. Nine minutes after actuation the safety injection system was secured.

October 21, 1973

On October 21, 1973, unit load was being gradually reduced from 450 MWe to perform plant maintenance when a turbine trip and resulting reactor trip occurred. At that time, the feedwater regulating system was programmed to open the regulating valves to 80% open on any trip. This resulted in a rapid filling of the steam generators and a cooldown of the RCS from 548°F to 470 F in about eight (8) minutes. Initiation of Safety Injection at 1685 psig terminated the event. As a result of this event, the feedwater regulating system was reprogrammed to provide 5% flow on a reactor trip, thereby preventing a recurrence of this event.

September 3, 1981

On September 3, 1981, with the unit operating at 390 MWe, a failure in the #1 Regulated Power Supply caused several alarms and the loss of several plant parameter indications. As a result, the operator manually tripped the plant, but feedwater flow continued, resulting in an overfilling of the steam generator. This resulted in RCS temperature falling from 550°F to 480°F and pressure falling from 2077 to 1700 psig in about five (5) minutes. Safety Injection which terminated the event was automatically initiated at the new setpoint of 1735 psig. As a result of this event, Westinghouse was consulted about the cooldown of the vessel. They confirmed that vessel shock was not a concern in this event.