Southern California Edison Company

P. O. BOX 800
2244 WALNUT GROVE AVENUE
ROSEMEAD, CALIFORNIA 91770

W. C. MOODY
MANAGER, NUCLEAR LICENSING

TELEPHONES
(213) 572-1817
(213) 572-1806

November 4, 1981

Director of Nuclear Reactor Regulation
Attention: D. G. Eisenhut, Director
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Gentlmen:

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

Your letter dated August 21, 1981 identified eight plants, including San Onofre Unit 1, from which additional information was required regarding their reactor vessels. This plant specific information would allow the NRC staff to evaluate the plants identified with respect to the subject issue.

Provided in the enclosure is a portion of that information requested within 60 days of your August 21, 1981 letter. As you know by our letter dated October 5, 1981 the response to Item 3, which required a limiting value for RT(NDT) for continued operation, has been delayed until completion of the Westinghouse Owners' Group work currently scheduled to be completed by the end of 1981. Our October 5, 1981 letter also indicated that the response to the remaining four items would be delayed an additional 15 days. The enclosure provides the responses to those four items.

In addition, your August 21, 1981 letter requested information within 150 days which addresses any necessary remedial or corrective action and responds to seven additional items requested by your letter. At the present time, we expect to provide to you the information within the 150 days (i.e., by January 18, 1982).

A001
S
1/1

If you have any questions regarding this information, please let me know.

Subscribed on this 4th day of November, 1981.

By W C Moody
W. C. Moody
Manager, Nuclear Licensing

Subscribed and Sworn to before me on this 4th day of November, 1981

Sheila N. Gauthier
Notary Public in and for the County of Los Angeles, State of California

OFFICIAL SEAL
SHEILA N. GAUTHIER
NOTARY PUBLIC-CALIFORNIA
PRINCIPAL OFFICE IN
LOS ANGELES COUNTY
My Commission Expires Nov. 4, 1983

Enclosure

ITEM 1

Provide the RT (NDT) values of the critical welds and plates (or forgings) for: a) initial (as-built) conditions and location (e.g., 1/4T) and b) current conditions (include fluence level) at the RPV inside carbon steel surface.

RESPONSE

The critical plates for the San Onofre Unit 1 reactor vessel are the intermediate shell plates identified as W7601-1, W7601-8 and W7601-9. Information regarding these plates is provided in WCAP 9520, "Analysis of Capusle F From the Southern California Edison Company San Onofre Reactor Vessel Radiation Surveillance Program" submitted to the NRC by letter dated April 18, 1980. A review of that document, in particular Table 5-3 indicates that the 30 ft-lb transition temperature increase for the plate material is significantly less than would be predicted by Regulatory Guide 1.99. A comparison of the transition temperature increases with predictions (see Figure 1) from a proposed ASTM Standard "Predicting Neutron Radiation Damage to Reactor Vessel Materials" shows much closer agreement. As a result, the .20 Cu ASTM curve was used to predict the current RT (NDT) for the intermediate shell plates. The change in the RT(NDT) for the plates is 162°F at the vessel surface and 123°F at 1/4 thickness (See Table 1).

The critical weld for the San Onofre Unit 1 reactor vessel is the intermediate longitudinal weld indentified as 7-860A. The surveillance weldment identified in WCAP 9520 was obtained from a longitudinal weld seam in the nozzle shell course of the vessel. It cannot be confirmed that the same heat of weld wire and lot of flux was used in the longitudinal welds of the intermediate region. Earlier efforts to determine the material content and properties of the beltline region welds have been documented in letters to the NRC dated November 10, 1977 and July 13, 1979. The November 10, 1977 letter was a response to an NRC request regarding the San Onofre Unit 1 Vessel Material Surveillance Program. The July 13, 1979 letter was a response to IE Bulletin 78-12 "Atypical Weld Material in Reactor Pressure Vessel Welds." In both letters it was indicated that documentation regarding the beltline region welds was not available. It was pointed out in the July 13, 1979 letter that the possibility of the use of atypical weld material on the vessel was remote since the vessel manufacturer utilized proper welding codes and no atypical weld material has been identified by the manufacturer in other vessels manufactured at the time of the San Onofre Unit 1 vessel. As a result of this situation it was conservatively assumed that the intermediate longitudinal welds contain .35% copper as opposed to .19% copper contained in the nozzle longitudinal welds. These welds are also considered to be low nickel (0.20% Ni) since the welding practice at the time of vessel fabrication resulted in low nickel welds.

Low nickel materials tend to show transition temperature increases significantly smaller than high nickel materials, as confirmed by recent NRC presentations which show that Regulatory Guide 1.99 overpredicts transition temperature increases by 50% for low nickel, high copper welds. As a result the .35% copper upper limit curve of Regulatory Guide 1.99 was adjusted by the 50% margin (see Figure 1) and was used to predict the current RT (NDT) of the limiting vessel weld (see Table 1.)

ITEM 2

At what rate is RT (NDT) increasing for these welds and plate material?

RESPONSE

Based on the information provided in Figure 1 and Table 1 the rate of RT(NDT) increase for the plate and weld material is as follows:

Intermediate Shell Plates	~4°F/EFPY
Intermediate Shell Longitudinal Welds	~3°F/EFPY

ITEM 3

What value of RT (NDT) for the critical welds and plate material do you consider appropriate as a limit for continued operation?

RESPONSE

As indicated in our October 5, 1981 letter a limit on RT (NDT) will be provided upon completion of the Westinghouse Owners Group work. This work is presently scheduled to be completed during December 1981. Based on the information provided in Items 1 and 2 and the conservative assumptions utilized the conclusions regarding continued operation of San Onofre Unit 1 as presented in our May 22, 1981 letter to the NRC are still applicable. San Onofre Unit 1 can safely sustain severe thermal shock transients, including repressurization, to beyond January, 1983.

ITEM 4

What is the basis for your proposed limit.

RESPONSE

RT (NDT) should not be utilized as a sole parameter to determine the acceptability of the integrity of the reactor vessel for any specific plant. RT (NDT) also should not be used as the sole parameter to compare the relative acceptability of different vessels. However, if a limiting RT (NDT) is to be defined for a specific vessel it should be qualified to the specific methodology utilized to calculate an acceptable lifetime for the specific vessel. The basic methodology that will be utilized in calculating an acceptable lifetime for a specific vessel is given in the report "Thermal Shock to Reactor Pressure Vessel" transmitted by the Westinghouse Owners Group to the NRC by letter dated May 14, 1981. This report describes, in general, the various operational and non-operational transients considered, the thermal and hydraulic methods, the fracture mechanics methods and the acceptance criteria. Using this methodology, an acceptable specific vessel life time can be determined. In addition to the Owners' Group work, plant specific analyses will be performed to evaluate the impact of key plant specific parameters such as fluence, transient characteristics, material properties, vessel geometry and weld locations on the acceptable vessel lifetime for San Onofre Unit 1. An RT (NDT) value for the vessel 1/4 thickness depth or any other vessel thickness depth can then be calculated for the acceptable vessel lifetime.

ITEM 5

Provide a listing of operator actions which are required for your plant to prevent pressurized thermal shock and to ensure vessel integrity. Include a description of the circumstances in which these operator actions are required to be taken. Included in this summary should be the specific, pressure temperature and level values for: a) high pressure injection (HPI) termination criteria presently used at your facility, b) HPI throttling criteria and instruction presently used at your facility, and c) criteria for throttling feedwater presently used at your facility. For each required operator action, give the information available to the operator and the time available for his decision and the required action. State how each required operator action is incorporated in plant operating procedures and in training and requalification training programs.

RESPONSE

No operator actions beyond those prescribed in existing procedures have been identified to date as being required to prevent pressurized thermal shock and to ensure vessel integrity for San Onofre Unit 1. This is based on the vessel integrity evaluations and results in the Westinghouse Owners' Group (WOG) report entitled "An Assessment of Westinghouse PWR Vessel Integrity for Severe Thermal Shock Conditions" submitted to the NRC by letter dated May 14, 1981.

The events of concern that produce the limiting severe thermal shocks relative to reactor vessel integrity include loss of coolant accidents and steam line breaks. The WOG Reference Emergency Operating Instructions developed since the TMI event were formulated considering the need for operator actions to ensure reactor vessel integrity. These guidelines have been incorporated into San Onofre Unit 1 Emergency Operating Instruction S-3-5.5 "Loss of Coolant". The operator actions prescribed in EOI S-3-5.5 relate to termination of safety injection and control of feedwater to prevent overcooling and represurization.

In the case of a loss of coolant accident the following SI termination criteria are prescribed:

1. Reactor coolant pressure is 2,000 psig and increasing AND

2. Pressurizer water level is greater than 50% of span, AND

3. Water level in at least one steam generator is stable and increasing as verified by level and by auxiliary feedwater flow to that steam generator. Total auxiliary feedwater flow to all steam generators should be greater than 200 gpm until indicated level is returned to within the narrow range level recorder AND

4. Reactor coolant indicated subcooling is greater than 40°F.

In the case of a steam line break the following SI termination criteria are prescribed:

1. T_H as determined by core exit thermocouples is less than 350°F AND

2. Wide range reactor coolant pressure is greater than 700 psig and is stable or increasing AND

3. Pressurizer water level is greater than 20% of span and rising (heaters covered) AND

4. The reactor coolant indicated subcooling is greater than 40°F AND

5. At least 200 gpm of auxiliary feedwater is injected into the steam generator(s) OR indicated narrow range water level in at least one steam generator is greater than 10 percent of span.

It should be noted that the SI system at San Onofre Unit 1 uses the feedwater pumps which have a shutoff head of 1160 psig so that the RCS cannot be repressurized above this value with the SI system operating.

There are no provisions for operator action to throttle SI in these events. In event of a loss of secondary coolant, the operator is initially instructed to throttle auxiliary feedwater flow to control runout and maintain greater than 200 gpm total auxiliary feedwater flow to the steam generators. After the water level in at least one steam generator has been restored to the narrow range span, the auxiliary feedwater flow is regulated to maintain an indicated narrow range water level or indicated wide range water level sufficient to assure that the steam generator feedrings are covered.

Information available to the operator so that appropriate operator action may be taken to terminate SI or throttle auxiliary feedwater include:

Pressurizer Pressure
Pressurizer Level
Containment PRessure
Containment Radiation
Containment Sump Level
Core Exit Thermocouples
Steamline Pressure
Steamline Flow
Steam Generator Water Level
Charging Pump Flow
Auxiliary Feedwater Flow
Main Feedwater Flow
Saturation Meter

The time available for the operator to take action to terminate safety injection or throttle auxiliary feedwater is dependent on the specific characteristics of the event. However, in general, operator action is not required or credited prior to 10 minutes following initiation of the event.

Training programs are established to address the procedural requirements for operator response during transient and accident conditions and the basis for these requirements. These programs provide through classroom review and simulator training of accident sequences, adequate opportunity for the operators to be cognizant of expected plant response and operator actions throughout any of the accidents that could adversely affect vessel integrity.

TABLE 1

SAN ONOFRE UNIT 1 REACTOR VESSEL

Component	Plate or Seam No.	Cu (%)	P (%)	Ni[a] (%)	Initial RT(NDT) (°F)	Current RT(NDT)[d] Inner Surface Fluence (10^{19} n/cm^2)	Current RT(NDT)[d] Inner Surface RT(NDT) (°F)	Current RT(NDT)[d] 1/4 Thickness Fluence (10^{19} n/cm^2)	Current RT(NDT)[d] 1/4 Thickness RT(NDT) (°F)
Inter Shell Plate	W7601-1	.17	.013	<.20	60	3.33	222[b]	1.40	183[b]
Inter Shell Plate	W7601-8	.18	.012	<.20	40	3.33	202[b]	1.40	163[b]
Inter Shell Plate	W7601-9	.18	.014	<.20	55	3.33	217[b]	1.40	178[b]
Inter Shell Long Weld Seam	7-860A	.35[a]	-	<.20	0	2.75	229[c]	1.16	192[c]

(a) Conservative Estimate - No Analysis Available.

(b) Based on slope of prediction curves presented in proposed ASTM Standard "Predicting Neutron Radiation Damage to Reactor Vessel Material"

(c) Based on Reg. Guide 1.99 Upper Limit Curve Reduced by 50% to Account for Low Nickel Content.

(d) San Onofre Unit 1 has operated for 8.93 EFPY as of October 31, 1981.

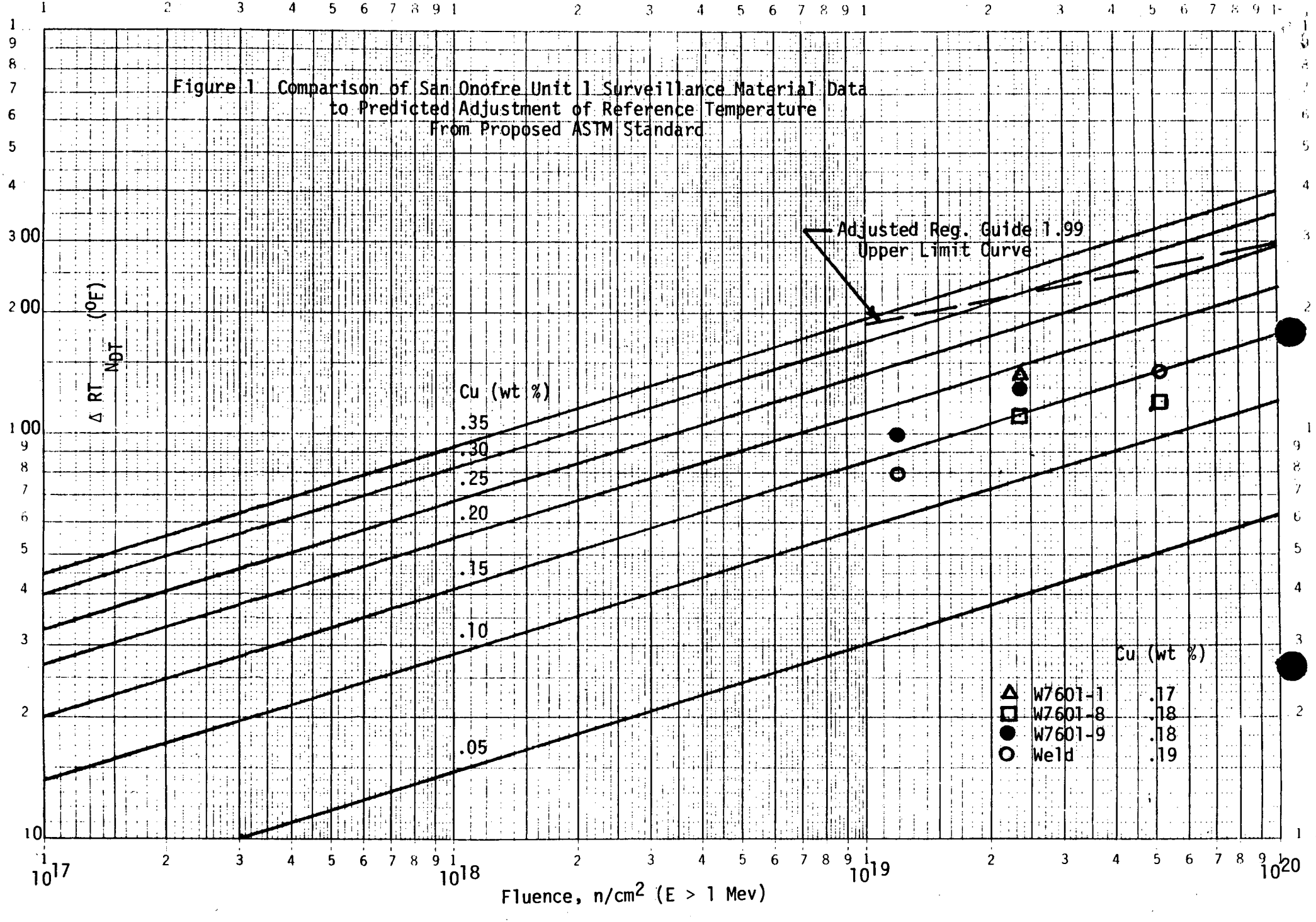

Figure 1 Comparison of San Onofre Unit 1 Surveillance Material Data to Predicted Adjustment of Reference Temperature From Proposed ASTM Standard