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Revision 1

EVALUATION OF THE PRIMARY SYSTEM PRESSURE BOUNDARY FOR SAN ONOFRE NUCLEAR GENERATING STATION UNIT 1 FOLLOWING A POSTULATED ATWS EVENT

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EXECUTIVE SUMMARY

This report has been prepared to evaluate the impact of a postulated anticipated transient without scram (ATWS) on the San Onofre Nuclear Generating Station Unit 1 primary system. The transient which produces the maximum pressure for the primary system was determined to be the loss of normal feedwater ATWS event, which results in a maximum pressure of 2998 psig at 700°F.

The evaluations reported here have used a consistent methodology, and shown that all the systems and components are capable of withstanding the transient.

TABLE OF CONTENTS

1.0 INTRODUCTION

According to the requirements of the Code of Federal Regulations, 10 CFR Part 50.62, each pressurized water reactor must have equipment to automatically mitigate the effects of an anticipated transient without scram (ATWS). In addition to the installation of diverse actuation circuitry to provide this mitigation capability, the NRC staff requested [1] that a demonstration be provided of the integrity of the primary coolant system pressure boundary and functionality of the valves needed for long term cooling following specified ATWS events.

The purpose of this report is to show that the San Onofre Nuclear Generating Station Unit 1 satisfactorily meets the ATWS criteria of NUREG-0460 [1], in that the structural integrity of the primary pressure boundary and the functionality of the valves needed for long term cooling will not be affected by the postulated ATWS event.

This demonstration of structural integrity has been completed on a generic basis for Westinghouse plants in earlier submittals, e.g. [2], but this report provides results specific to San Onofre Unit 1. San Onofre Unit 1 was designed to Section VIII of the ASME Code, except for piping components which were designed to ASA B31.1 Code.

1.1 Development of Limiting Transient

The first step in performing this evaluation was to develop the most limiting ATWS transient for San Onofre Unit 1, and this was accomplished in reference 3, where a number of candidate events were developed in this report. Of these events, the one which is most limiting with regard to reactor coolant system pressure is the loss of normal feedwater ATWS event.

A loss of normal feedwater could result either from a malfunction in the feedwater condensate system or its control system from such causes as simultaneous trip of both condensate pumps, simultaneous trip of both main feedwater pumps (or closure of discharge valves), or simultaneous closure of all feedwater control valves. The vast majority of these cases would cause only a partial loss of feedwater flow.

A loss of main feedwater produces a large imbalance in the heat source/sink relationship. When feedwater flow to the steam generators is terminated, the secondary system can no longer remove all of the heat that is generated in the reactor core. This heat buildup in the primary system is indicated by rising reactor coolant system temperature and pressure, and by increasing the pressurizer water level, which is due to the insurge of expanding reactor coolant. Water level in the steam generators falls as the remaining water in the secondary system, un-replenished by main feedwater flow, is boiled off. When the steam generator water level falls to the point where the steam generator tube bundle is exposed and primary-to-secondary heat transfer is reduced, the reactor coolant temperature and pressure begin to increase at a greater rate. This greater rate of primary system temperature and pressure increase is maintained as the pressurizer fills and discharges water through the safety and relief valves. Reactivity feedback, due to the high primary system temperature, reduces core power. Eventually, the system pressure begins to decrease, and a steam space is again formed in the pressurizer.

For the Loss of Normal Feedwater event, a turbine trip signal is assumed to result from the ATWS Mitigation System Actuation Circuitry (AMSAC). Since the reactor coolant pumps (RCPs) trip on the turbine trip signal, a three-stage RC pump coastdown sequence assumed for the Loss of Load event was assumed in this analysis. Specifically, following the AMSAC, the RCPs continue to run at full speed for one minute and are powered from an offsite grid. After one minute at full speed, the RCPs were modeled to coastdown with the turbine generator for 4 minutes until the RCP breakers open at 40% pump speed. At this time (5 minutes after AMSAC), the RCPs coastdown with the flywheel.

Plant behavior was evaluated for a loss of normal feedwater event occurring from full power with the assumption that the control rods fail to drop into the core, or there is no generation of a reactor trip signal. The evaluation shows the effectiveness of the reactor coolant system pressure-relief devices to reduce the magnitude of the reactor coolant system pressure transient.

The following assumptions were made in the analysis:

- No credit for automatic reactor trip

- Initial normal full power operation early in core life. Since negative reactivity feedback is essential to reduce core power for ATWS events, early in core life conditions are conservatively assumed since the core becomes inherently more negative with core life.

- A complete loss of main feedwater event occurs at t = 0. Main feedwater falls to zero in the first four seconds of the transient with no main feedwater after this time.

- No credit for automatic control rod insertion as reactor coolant temperature rises.

- No credit for pressurizer sprays in reducing pressure.

- Pressurizer pressure relief through all power-operated and spring loaded relief valves.

- The AMSAC signal is actuated when the steam generator water level reaches 5% narrow range span.

- An AMSAC actuated turbine trip occurs at 30 seconds after the AMSAC setpoint is reached.

- The reactor coolant pumps trip on the AMSAC actuated turbine trip signal and coastdown as earlier described.

- AMSAC actuated auxiliary feedwater at 100°F begins at 60 seconds after the AMSAC setpoint is reached at a total flow rate of 185 gpm, delivered equally to all three steam generators. Auxiliary feedwater is not delivered until a volume of 73 ft^3 of hot main feedwater at approximately 400°F has been purged from the main feedwater lines.

- o Primary to secondary heat transfer area is reduced as the steam generator shell-side water inventory falls below the value necessary to cover the tube bundle.

- o Steam dump control system is not modeled.

- o Plant operation at nominal T_{avg} = 551.5°F

Figures 1-1 through 1-9 show the plant transient response for a loss of normal feedwater without reactor trip for SONGS-1. The sequence of events for this transient are shown in Table 1-1.

The peak pressure in the reactor coolant system is 2998 psi and occurs at 102 seconds after initiation of the event for the case assuming nominal full power initial conditions. The pressurizer reaches a peak pressure of 2942 psi at the same time, while relieving approximately 560 lb/sec of water.

At ten minutes into the transient, conditions are stabilized, with auxiliary feedwater providing heat removal capability and with an intact reactor coolant system and core. Thus, the operator could begin shutdown operations through rod insertion, actuation of the safety injection system, or through the Borate or Emergency Borate modes of the Chemical and Volume Control System.

TABLE 1-1
SEQUENCE OF EVENTS FOR THE
LOSS OF NORMAL FEEDWATER ATWS EVENT

Event	Time (seconds)
Loss of main feedwater begins	0
Main feedwater flow completely lost	4
Pressurizer relief valves lift	23
Low SG level AMSAC setpoint reached	33
Steam generator safety valves lift	53
Turbine trip via AMSAC signal	63
Initiation of RCP trip sequence	63
Pressurizer safety valves lift and pressurizer fills with water	90
Auxiliary feedwater pumps start via AMSAC signal	93
Maximum RCS pressure reached	102
Reactor coolant pump coastdown with generator begins	123
RCP speed falls to 40% and inertial coastdown with flywheel begins	363

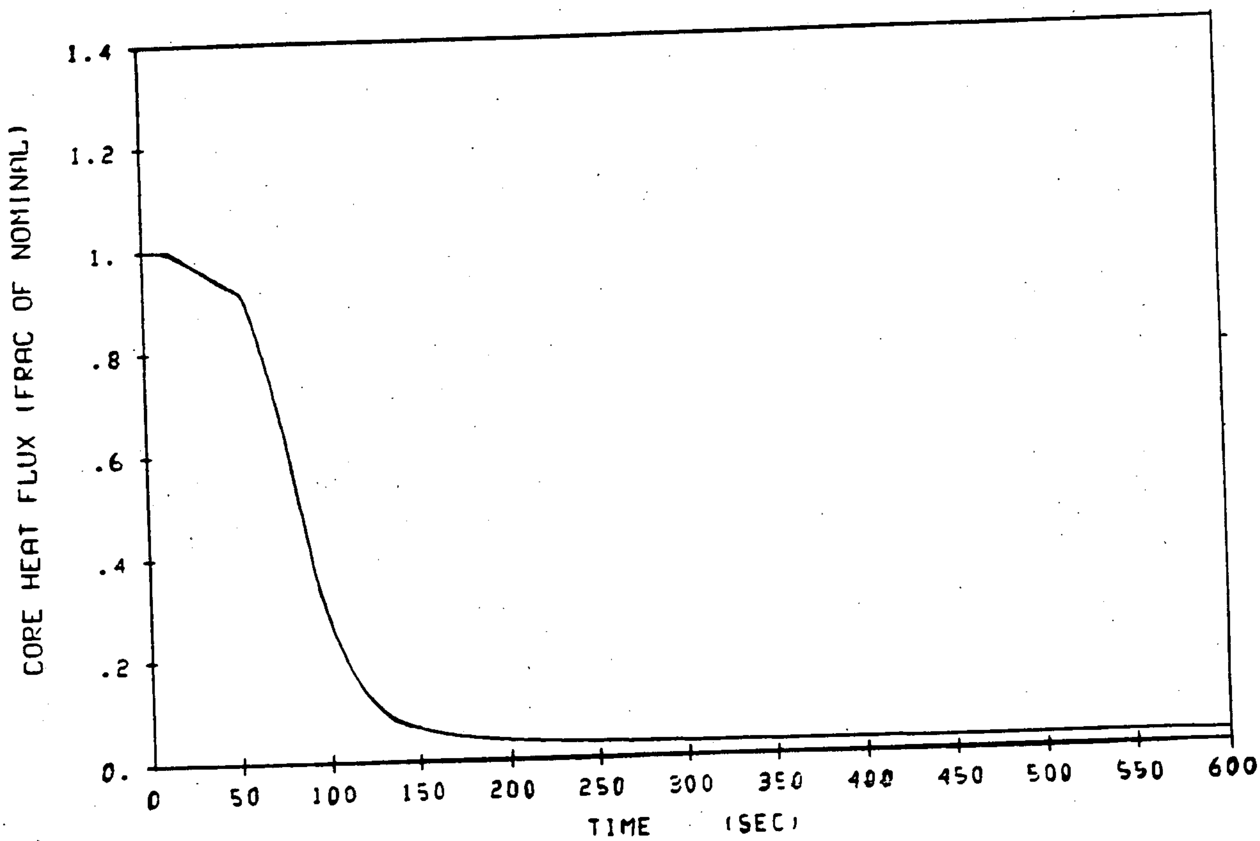

Figure 1-1. Core Heat Flux versus Time Loss of Normal Feedwater ATWS Event

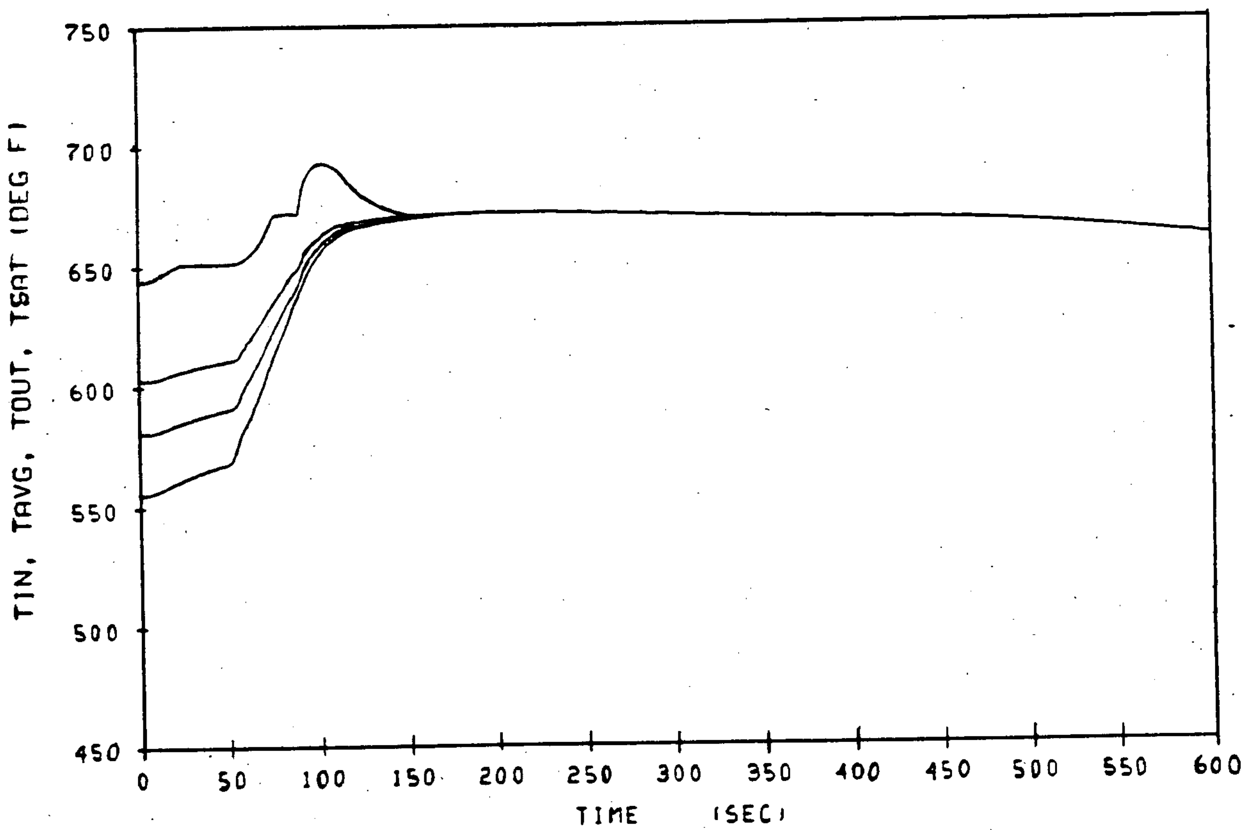

Figure 1-2. Reactor Coolant Temperatures versus Time Loss of Normal Feedwater ATWS Event

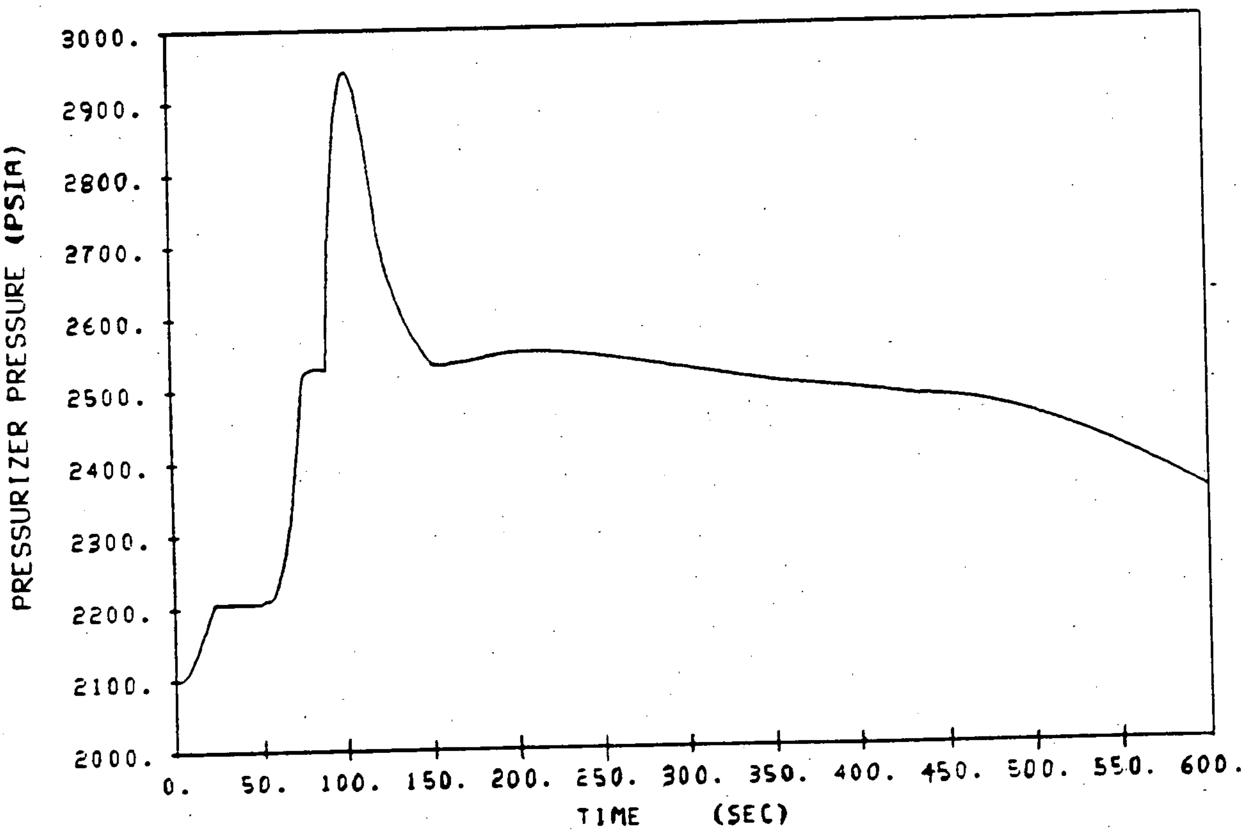

Figure 1-3. Pressurizer Pressure versus Time Loss of Normal Feedwater ATWS Event

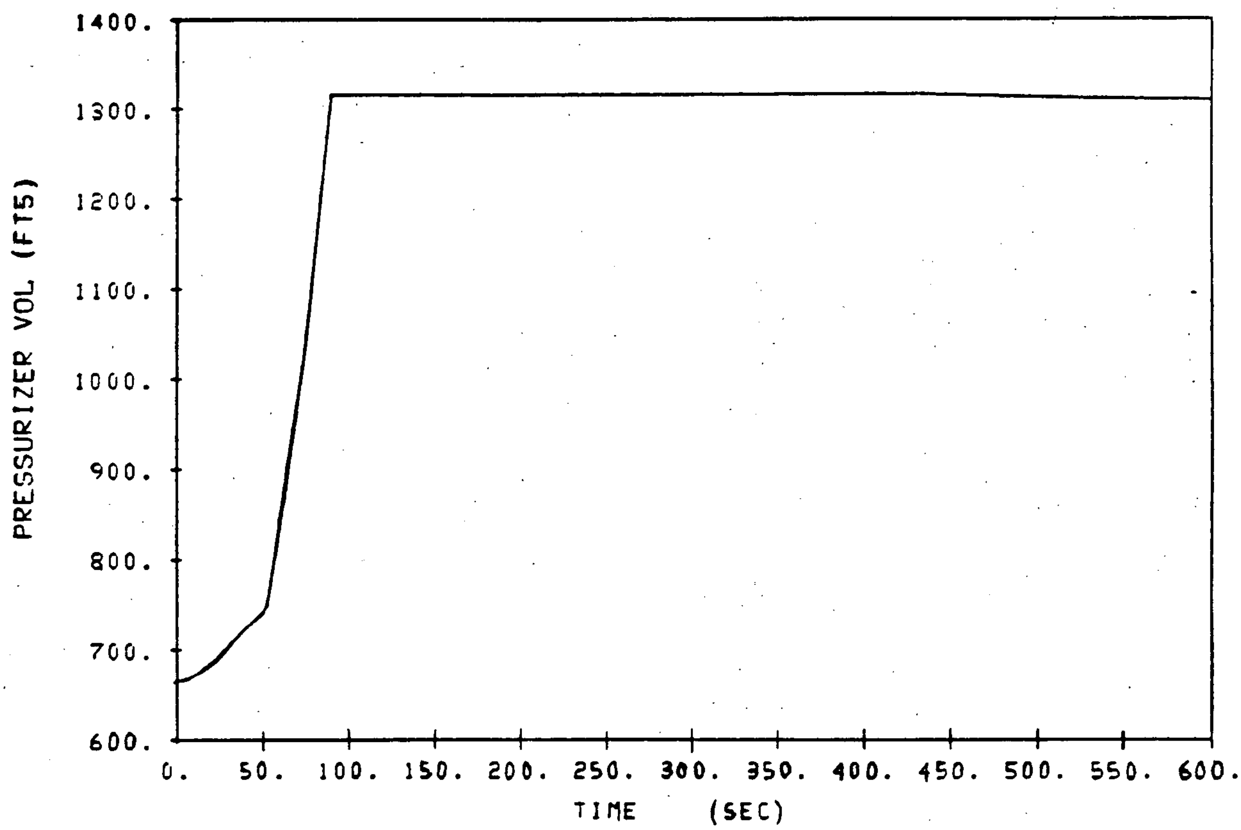

Figure 1-4. Pressurizer Volume versus Time Loss of Normal Feedwater ATWS Event

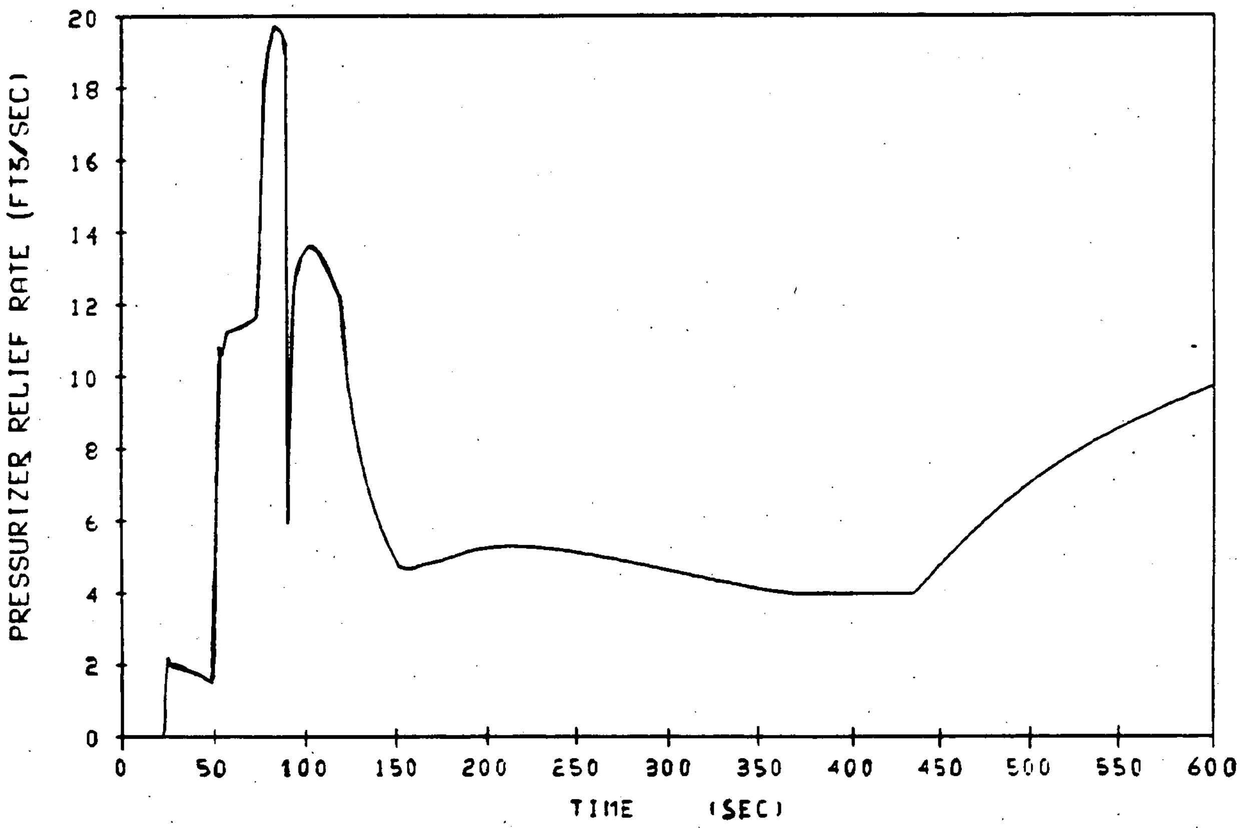

Figure 1-5. Pressurizer Relief Rate versus Time Loss of Normal Feedwater ATWS Event

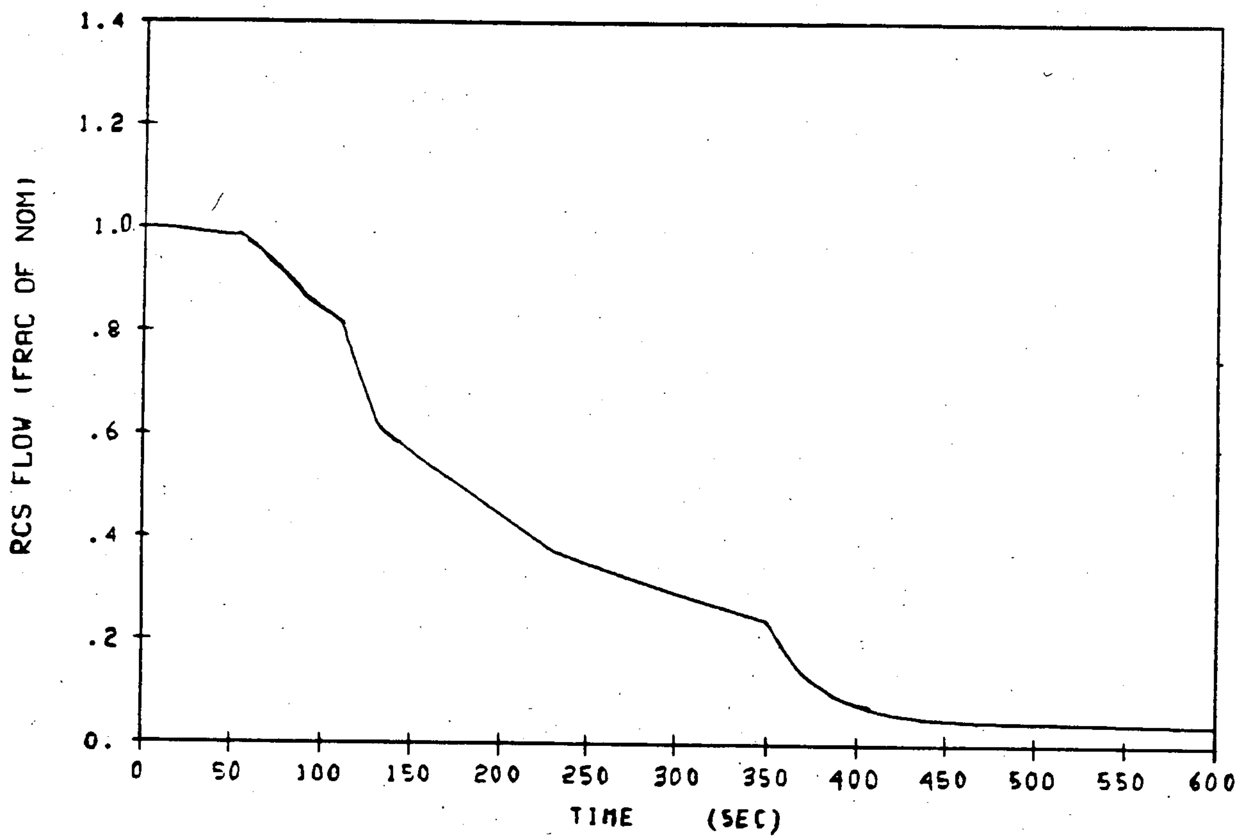

Figure 1-6. Reactor Coolant Flow versus Time Loss of Normal Feedwater ATWS Event

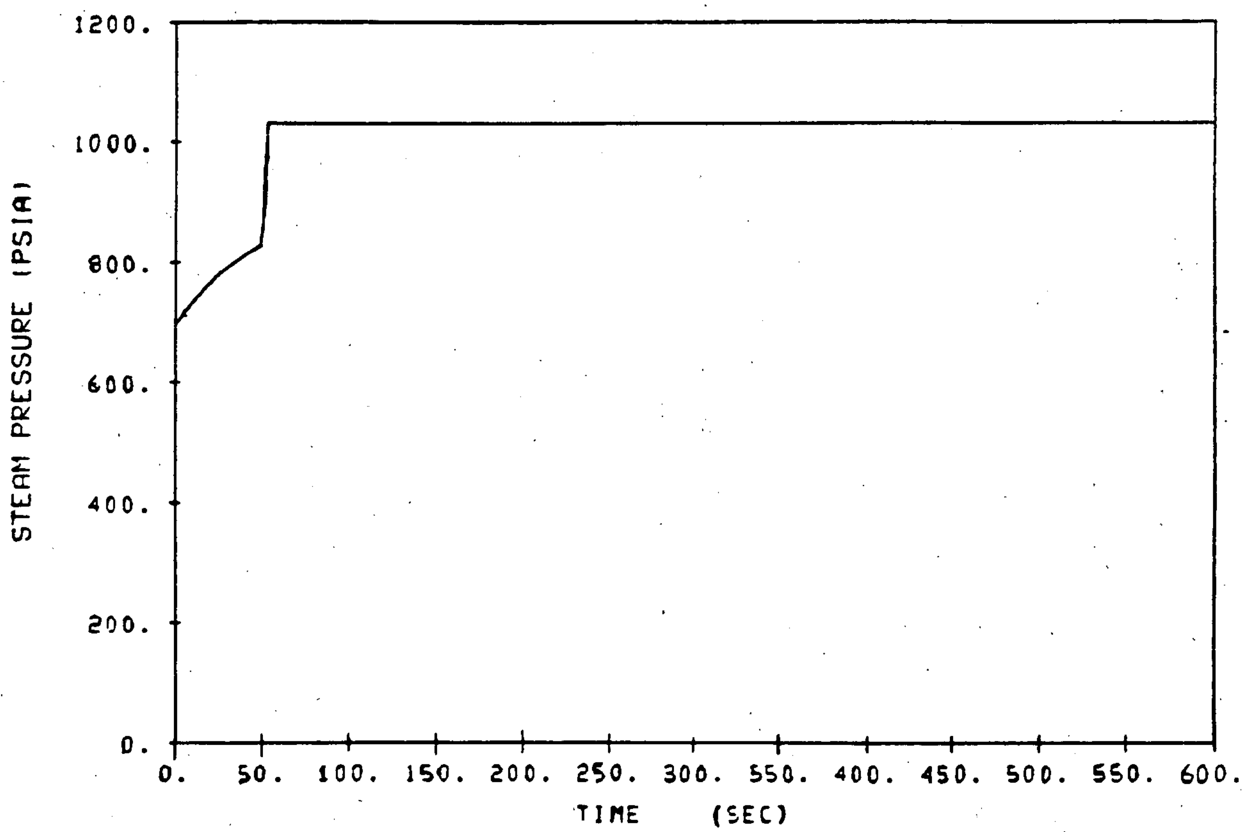

Figure 1-7. Steam Pressure versus Time Loss of Normal Feedwater ATWS Event

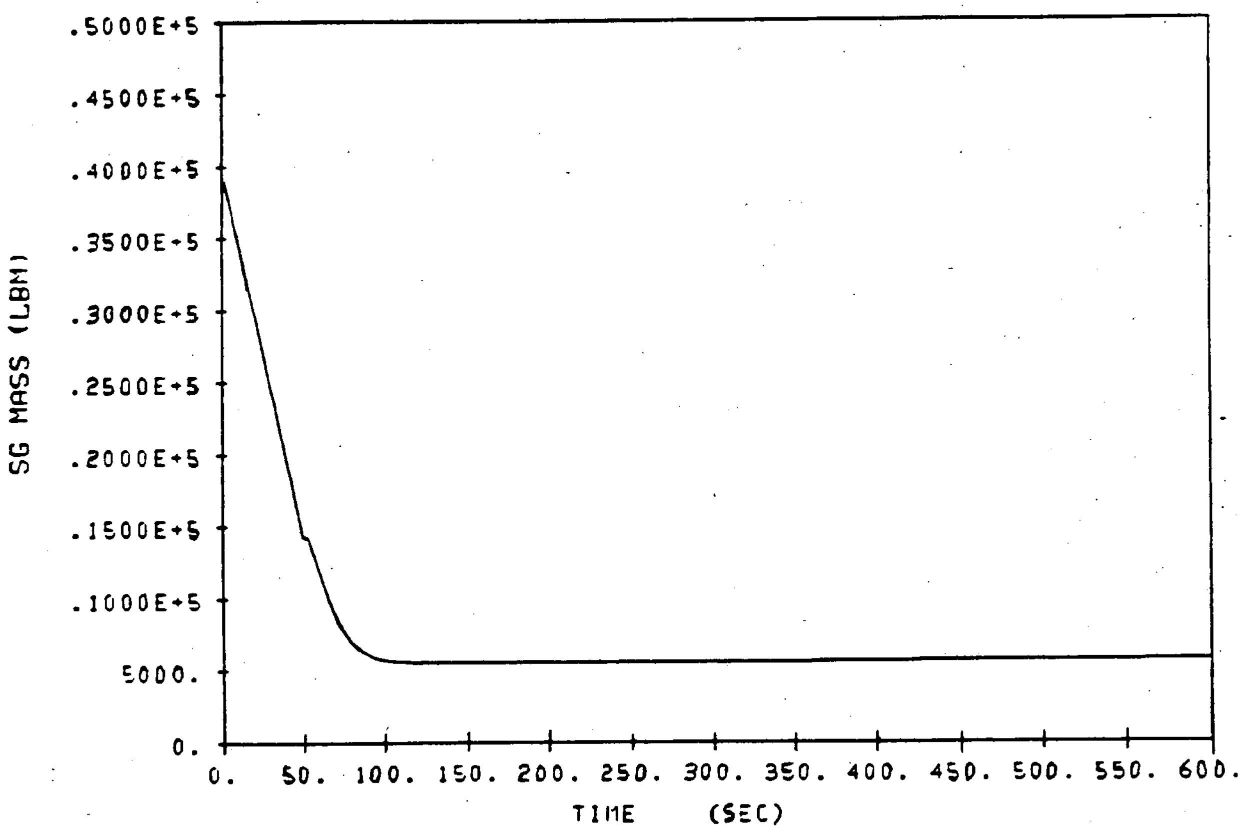

Figure 1-8. Steam Generator Mass versus Time Loss of Normal Feedwater ATWS Event

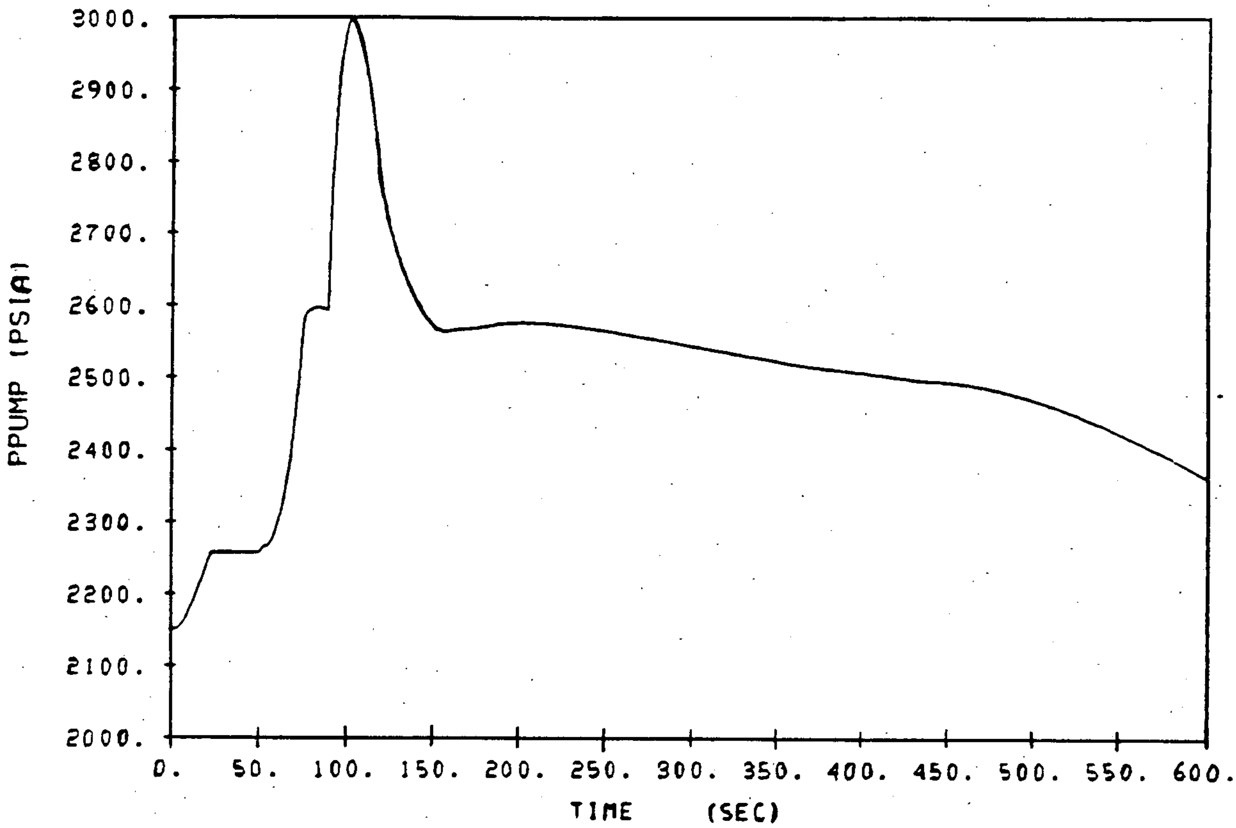

Figure 1-9. Reactor Coolant System Pressure versus Time Loss of Normal Feedwater ATWS Event

2.0 OVERALL APPROACH

The San Onofre Unit 1 reactor coolant system and components were designed to Section VIII of the ASME Code (ASA B31.1 for the piping), which does not require a fatigue analysis. Because of the vintage of the plant, detailed stress analyses are not available for all the components. Therefore, rather than perform detailed analyses of the various components to compare directly with the allowables of the Code, the following conservative approach was utilized.

The Anticipated Transient Without Scram evaluated herein can be considered as a hydrotest conducted at a higher temperature, but lower pressure, than the original preservice hydrotest. The original preservice hydrotest was conducted at 3728 psig, and at room temperatures, which is intuitively a much more severe test of system integrity than the ATWS, which is assumed to occur at 700°F, and reaches a maximum pressure of 2998 psi (2984 psig). The overall approach adopted for this report is to demonstrate this quantitatively for each component.

It is assumed in each of the analyses to follow that the stresses in each component are acceptable for the preservice hydrostatic test, at 3728 psig. This can be a very conservative assumption, since in many locations the stresses will be only a small fraction of the allowables, as seen for example in the detailed comparisons provided in the earlier generic submittal [2].

If the elastically calculated stress is assumed to be at the allowable for the hydrotest, it will be less than the allowable by the ratio 2984/3728 for the ATWS transient. This is the calculation method used to evaluate each of the components to be discussed in the sections which follow.

Since the ATWS event is conservatively assumed to occur at a temperature of 700°F, the allowable stress for the event must therefore be reduced to account for the higher temperature. The magnitude of this reduction will be different for different materials, with larger effects being seen for the austenitic stainless materials, and the smallest effects on the Alloy 600 components.

This approach has been used for each of the components, and the results for each component are discussed in the sections which follow. In each case, except the steam generator and piping, Section VIII allowables and material properties have been used, since that is the code of record for the plant. For the piping, ~~ASA~~ B31.1 allowables were used, and for the steam generator, Section III (1965) allowables were used. The ATWS event evaluated herein has been assumed to be independent of seismic and other operational transients.

ASME.

3.0 REACTOR PRESSURE VESSEL AND CONTROL ROD DRIVE MECHANISMS

3.1 Reactor Vessel

The San Onofre 1 reactor vessel was designed and constructed to the requirements of the 1959 Edition of Section VIII of the ASME Boiler and Pressure Vessel Code with the appropriate "nuclear" code cases. As a result the reactor vessel does not have a detailed ASME Section III stress report, since Section VIII requires only basic sizing calculations. However, these basic sizing calculations using the Section VIII, Div. 1 allowable stress values provide the necessary assurance that the primary stress intensities, due to design operating conditions and hydrostatic test, remain at levels at which the integrity of the vessel is maintained.

The ATWS evaluation for the reactor vessel is performed on a general basis considering the allowable stresses for the various vessel materials, the ASME Section VIII hydrostatic test at 3750 psi reconciled by the sizing calculations. Using these considerations the maintenance of structural integrity of the reactor vessel for the ATWS conditions of 2998 psi maximum pressure at 700°F maximum temperature is demonstrated.

Table 3-1 provides a list of all of the reactor vessel pressure boundary materials with a ratio of the design allowable stresses at 700°F versus the design allowable stress at the 70°F ambient temperature for the hydrostatic test at 3750 psi. Based on proportionality calculations using these ratios times the 3750 psi hydrostatic test pressure, all parts of the reactor vessel pressure boundary, with the exception of the bolting, are acceptable for the ATWS pressure of 2998 psi. Using this proportionality method, the maximum allowable pressure for ATWS, excluding the bolting, is approximately 3170 psi. Therefore, all parts of the vessel pressure boundary, excluding bolting, are acceptable for ATWS with margin for additional external loadings.

The reactor vessel bolting material for the closure studs and nuts is high strength low alloy steel bolting in accordance with a "nuclear" code case which was in effect at the time that the San Onofre 1 reactor vessel was being designed and manufactured. The AISI 4340 material to the requirements of SA-193 is not included in the current listing of materials and allowable stresses in ASME Section VIII, Division 1. The maximum allowable stress intensity values used in the design of the San Onofre 1 reactor vessel studs and nuts are found in Document PB-151987 [16] which was issued by the U.S. Department of Commerce in December 1958. This document lists the primary plus secondary stress intensity limit S_p at 70°F as 90 ksi while the S_p at 700°F is listed as 80 ksi. These values provide a ratio of 0.89 for demonstrating the ATWS structural integrity. Scaling the 3750 psi hydrostatic test pressure using this ratio yields a pressure of 3330 psi. This permissible ATWS pressure for the reactor vessel bolting covers the 2998 psi calculated maximum pressure with an 11 percent margin of safety.

3.2 CRDM Pressure Housing

This evaluation of the ATWS effect on the CRDM pressure housing is based on a comparison of the hydrostatic test at room temperature and the ATWS pressure at 700°F, with adjustment made for the temperature difference.

The CRDM pressure housing is made from SA-336-F8 and SA-193-B8 material--annealed 304 SST. The CRDM housings were hydrostatically tested at 6300 psig at room temperature. By determining the ratio of the yield stresses from 700°F to room temperature, we can project an equivalent hydro test at the elevated temperature. Although by using this ratio, it is assumed that the housings remain in the elastic range at pressure. It is acknowledged that this assumption is not entirely accurate, but the ratio method is a good approximation.

The equivalent hydrostatic pressure at 700°F is 3528 psig. Having established by test and analytical projection, that the CRDM housings structural integrity is sound at this pressure and temperature, the ATWS pressure of 2998 psi is considered safe for the CRDMs.

TABLE 3-1

REACTOR VESSEL PRESSURE BOUNDARY MATERIALS

Material	Parts	S 70°F (ksi)	S 700°F (ksi)	S(700°F) / S (70°F)
1. SA 302, GR.B	Shell and Heads	20.0	20.0	1.0
2. SA 336, Case 1236	Head Flange, Vessel Flange, Primary Nozzles	20.0	20.0	1.0
3. SB-167 (Alloy 600)	Head Adapter (CRDM Housing) Tubes	20.0	20.0	1.0
4. SA-182, F304	Head Adapter (CRDM Housing) Flanges	17.5	14.8	.85
5. SA-182, F316	Primary Nozzle Safe Ends	17.5	16.3	.93
6. SA 193 AISI 4340 (SA-540, Cl. 3)	Closure Bolting	90*	80*	.89

* S_p values from Reference 16

4.0 PRESSURIZER

An assessment was made of the pressurizer integrity under the anticipated transient without scram (ATWS) parameters of 2998 psig and 700°F. The assessment was made considering the ATWS transient as a Level C Condition and using the procedure outlined in Section 2. The assessment shows that for the most critical region (head region), the allowable pressure is 3536 psig, which is greater than the 2998 psi ATWS pressure. The acceptability of the ATWS transient is based on the assumption that the external nozzle loads during the ATWS transients are either enveloped by the design loads or are of such a magnitude that they can be accommodated by the margins available.

5.0 STEAM GENERATOR PRIMARY PRESSURE BOUNDARY

Steam generator primary pressure boundary components were evaluated for the ATWS maximum primary to secondary pressure difference of 1912 psi (2942-1030). The critical components evaluated were the tubing and tubesheet, using the corresponding material stress limits specified in the ASME Code for the Emergency condition (level C) loads. For SONGS-1, Section III limits* were used, although the steam generators were built to Section VIII. The method used is essentially that detailed in Section 2 of this report.

In this evaluation, the Emergency stress limits from Section III were used, which are:

1.2 Sm or Sy for Primary membrane stress (P_m), and
1.8 Sm for Primary (local) membrane plus bending stress ($P_m + P_b$, P_L).

According to the calculations, the steam generator tubing (SB-163, I-600 MA) was found to be the most limiting component in the steam generator primary pressure boundary. The ATWS transient pressure was shown to be acceptable for design conditions with the stress limit of 1.2 Sm and original tube thickness, which was conservatively assumed as drawing minimum minus 0.003" allowance for general errosion and corrosion. Similarly, for locally degraded tubing the ATWS pressure was shown to be acceptable by using the stress limit of Sy, which is conservative and used in Regulatory Guide 1.121 for upset condition evaluation. The tube thickness assumed in degraded condition is 40% of the original wall. This translates to 60% local degradation, which is the sum of the Tech. Spec. specified limit of 50% and 10% allowance for inspection uncertainty and degradation during the operating cycle.

The tubesheet (SA-266 Gr 2) is less limiting than the tubing and meets the ATWS transient pressure requirement (ΔP = 1912 psi) without compromising the primary stress limits.

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* The Sy limit, per Regulatory Guide 1.121, is used in the limiting tube pressure calculations. It is to be noted that these calculations do not constitute a complete set of Regulatory Guide 1.121 plugging criteria calculations.

6.0 PIPING

The piping evaluated for the ATWS transient includes the Reactor Coolant Loop (RCL) piping and the Auxiliary lines connected to the RCL piping and primary equipment. The RCL piping includes the hot, cold, and crossover leg for each loop, the elbows on each leg, and the auxiliary nozzles attached to the RCL piping. The primary equipment and valves are not evaluated as a part of the RCL piping and results for these components are reported separately in other sections.

The auxiliary lines included in this evaluation are those which are pressurized up to the ATWS pressure during the postulated transient. This pressurization typically occurs up to the second normally closed valve in the auxiliary line. Included in the auxiliary lines evaluation are elbows, tees, reducers, flanges, and other fittings and components which are a part of the line.

The piping evaluation performed by Westinghouse used a simplified method to calculate allowable ATWS pressure from a ratio of design allowable stress at ATWS temperature divided by allowable stress at 70°F, as discussed in Section 2. The equation used in the calculation of allowable ATWS pressure is as follows:

$$P_{(ATWS\ allow)} = \frac{S_{(ATWS\ temp)}}{S_{(70°F)}} \times 1.5 \times P_{(design)}$$

The allowable ATWS pressure was compared to the actual ATWS pressure of 2984 psig to determine acceptability.

The individual terms of the equation are defined as follows:

ATWS temperature is 700°F.

$P_{(design)}$ is the design pressure of the system which is equal to 2485 psi.

The allowable stresses(S) at 70°F and at ATWS temperature are taken from the 1989 version of the B31.1 ASME code, and are tabulated in tables 6-1 and 6-2.

Allowable pressure has been calculated for each material used in the fabrication of the RCL and auxiliary line piping systems. See tables 6-1 and 6-2.

Results of the evaluation show that the minimum allowable pressure for ATWS is 3150.0 psig, except for the 3/8" sample tubing. This is greater than the actual ATWS pressure of 2984 psi and is, therefore, acceptable.

Regarding the 3/8" sample tubing in the material specification, the grade of material has not been specified. Therefore, the allowable ATWS pressure has been calculated for each grade, and the material with the lowest allowable pressure is reported in table 6-2. The material is A 213, TP347 (with an allowable stress at 700°F equal to 14.7 ksi). Allowable stress for the sample tubing with material A 213, TP316L at 700°F is given as 14.7 ksi. Sample tubing stresses have been calculated with this assumed material, to show that allowable stress at the ATWS temperature has not been exceeded. All other grade materials give allowable ATWS pressure above 2984 psig except grade TP310. However, grade TP310 has an allowable stress at 700°F above that for TP347. Therefore, the sample tubing is shown to be acceptable for the ATWS pressure transient.

The piping systems can also be shown acceptable for the requirements of the ASME Code Section III (Winter 1980 edition).

TABLE 6-1
ALLOWABLE ATWS PRESSURE FOR RCL PIPING

Material	$S_{70°F}$ (ksi)	$S_{700°F}$ (ksi)	P_{allow} (psig)	Location
1. SA 351, Gr CF8M	14.0	13.0	3461	RCL elbows and fittings
2. SA 376, TP316	18.8	16.3	3232	RCL piping and surgeline
3. SA 182, F316	17.5	16.3	3472	RCL forgings
4. SA 403, WP316	18.8	16.3	3232	RCL fittings

TABLE 6-2

ALLOWABLE ATWS PRESSURE FOR AUXILIARY PIPING

Material	$S_{70°F}$ (ksi)	$S_{700°F}$ (ksi)	P_{allow} (psig)	Location
1. A 312, TP316	18.8	16.3	3232	Piping 1/2"
TP316L	15.7	12.9	3062	to 12"
2. A 182, F316	17.5	16.3	3472	Flanges & 2"
F316L	15.7	12.9	3062	& smaller fittings
3. A 403, WP316	18.8	16.3	3232	Fittings
WP316L	15.7	12.9	3062	2-1/2" to 12"
4. A 312, TP304	18.8	15.9	3152	2" SWI and
TP304L	15.7	13.9	3205	2" SI after first valve
5. A 182, F304	17.5	14.8	3152	Forged
F304L	15.5	13.1	3150	fitting on 2" SWI & 2" SI line
6. A 213, TP347	18.8	14.7	2914*	3/8" soft
TP316L	15.7	12.9	3062	annealed sample tubing

* Allowable pressure is less than ATWS pressure. Calculations of actual stress at ATWS pressure of 2984 psig, shows that actual stress is less than 12.9 ksi.

7.0 REACTOR COOLANT PUMP

7.1 Background

The reactor coolant pumps (RCP) were evaluated for their ability to accept a pressure of 2998 psi at 700°F on a simplified basis. The approach taken was to consider the capacity of the RCP to accept the 2998 psi pressure as a static hydrotest pressure at 700°F static loop temperature. The benchmark for this simplified approach is the demonstrated capability of the RCPs to accept the hydrostatic test pressure of 3750 psi required by the ASME Code requirements of the SCE Equipment Specification, 675199, Rev. 1. (This specification references ASME Section VIII -- no year specified -- and paragraph UG-99(c) as the applicable hydrotest requirement.)

The approach utilized herein is to define a "hydrostatic test capability" relative to the benchmark hydrotest. The intent here is solely to avoid exceeding the stress state established in the original equipment hydrotest; this is accomplished by reducing the pressure which is applied to the equipment in accordance with appropriate material property reductions due to the assumed increase in metal temperature from 70°F (original hydrotest temperature) to 700°F (ATWS peak temperature).

7.2 Results of Evaluation of Pressure Capacity of RCP at 700°F

The material properties to be utilized in this evaluation of the capability of the RCP are the "S" values in the ASME Code, Section VIII (similar to the "Sm" concept in the Section III Code). The percentage reduction in S due to increasing the temperature of the pump to 700°F from 70°F is a maximum for the bolting of the main closure and is 21 percent. The ATWS pressure (2998 psi) applied to the pump is reduced by 20 percent relative to the hydrotest pressure of 3750 psi, but the allowable is reduced 21 percent, which makes the maximum pressure for the postulated ATWS 35 psi higher than the allowable.

Therefore, the ATWS transient will pressurize the pump to a state which is slightly more severe than the original equipment hydrotest at 3750 psi, 70°F.

There is considerable conservatism in this calculated "hydrostatic test pressure" capability, however, simply because it assumes that all of the RCP components have reached 700°F during the ATWS transient. This is impossible, of course, since it would be necessary to completely insulate the pump from the surrounding air and allow no water flow across the RCP boundaries to accomplish this. Further, for any significant portion of the pressure boundary components to rise to within 50°F of the loop temperature it is necessary that the loop temperature remain at the specified temperature for a time period on the order of hours, not minutes as is typical of ATWS transients. Therefore, it is logical to conclude that if a thermal map of the pressure boundary parts were developed that for any short duration ATWS transient (say 15 minutes or less), the temperatures of the pressure boundary parts could be demonstrated to be far lower than the 700°F loop water (a reasonable and conservative value would be 665°F, based on figure 1-2); the reduction in material property values could then be shown to be less than the 20 percent reduction in pressure, thus rendering the pump acceptable for this transient on this basis.

7.3 RCP Seals

A final consideration is the capability of the RCP seals to accept this pressure/temperature. The pump seals, by virtue of the built-in protection from high temperature exposure afforded by either the injection water to the pump or by the heat exchanger under the pump bearing, are never exposed to temperatures above (nominally) 200°F in normal pump operation. With continued injection and cooling water flow into the pump, the seals would be exposed only to the pressure transient. It is expected that seal operation would continue normally if the pressure peaks momentarily at 2998 psi since the seal design has already demonstrated a capability of accepting a 3750 psi hydrostatic test pressure.

7.4 Conclusion

The conclusion from this evaluation is that the RCPs are acceptable for the ATWS pressure/temperature transient at 2998 psi/700°F solely on the basis of demonstrated hydrostatic test capability of the original equipment. The limiting issue in this evaluation is detailed knowledge of the temperature of the parts during the transient.

8.0 RCS PRESSURE BOUNDARY VALVES

To demonstrate structural integrity for RCS pressure boundary valves when subjected to an ATWS transient of 2984 psig at 700°F, the hydrostatic test pressure was assumed in accordance with B16.5 section 8.0 to be 1-1/2 times its system design pressure. Material allowables for valve bodies, body to bonnet bolting, and valve discs were obtained from Section VIII of the ASME Boiler and Pressure Vessel code. The valves considered in this analysis are listed in Table 8-1.

8.1 Valve Bodies and Discs

The system design pressure for the valves in the RCS pressure boundary was identified by the valve specification to be at least 2485 psig at 650°F. The valve discs and bodies were assumed to be hydrostatically tested at 1-1/2 times the system pressure which results in a hydrostatic test pressure of at least 3727 psig. To compensate for the reduction in material yield properties which will occur when the material temperature ranges from 100°F to 700°F, a ratio of design allowable strength for the various valve bodies and discs was calculated.

Based on this compensation for temperature the critical valve material was identified as SA182 F316L, the hydrostatic test results provide verification of structural integrity up to 3069 psig at 700°F.

8.2 Valve Bolting

As previously mentioned, all RCS pressure boundary valves are subjected to a hydrostatic shell test. For those valves employing a body to bonnet bolted flange joint (typically globe and gate valves) the hydrostatic shell test is sufficient to demonstrate structural integrity of the bonnet-body bolting. To compensate for the reduction in the material yield properties between 100°F to 700°F, a ratio of design allowable strength for the various bolting materials was calculated. Based upon this compensation for temperature the critical bolt material was identified as A-453 Gr 660, the hydrostatic test results provide verification of structural integrity up to 3526 psig.

8.3 Valve Function

The stresses in the valve disc resulting from the ATWS transient will not exceed the valve disc's material yield strength which does not affect the valve's ability to function after the ATWS transient.

TABLE 8-1 VALVES CONSIDERED

Valve Location	Valve ID	W Specification & Sheet	Vendor Drawing No.	Purpose
287	287	676044/5 2485 psig 650°F	B.S. & B 95215-15	Excess Letdown Isolation
305	305	676044/6 2700 psig 600°F	B.S. & B 95215-2	Charging return to Loop A Shut-off Valve
304	-	676044/6 Max. 2700 200 psig* 600°F	B.S. & B 95215-15	Charging return to Loop A Shut-off valve *1. Max. pressure is 2700 psig. 2. P&ID 540F275 show valve 308 after this valve Max. pressure for 30 is 2511 psig.
356 357 358	-	675198/2 2485 psig 650°F	Edward P-19341	Boric Acid Injection Line Motor Operated Block Valve to Loop A, B, C
522 523	3/4-XSBN	None/17 2511 psig 650°F	Whitey	Pressurizer Spray PCV Bypass (need valve)
530		676044/1G 0485 psig 680°F	Anchor Darling 10529	PORV Block Valve
532 533		675197	Crosby H-47469	Pressurizer Safety Valve

Note: 1. B.S. & B: Black Sivalls & Bryson

TABLE 8-1 VALVES CONSIDERED (cont'd)

Valve Location	Valve ID	W Specification & Sheet	Vendor Drawing No.	Purpose
545 546		676044/2 3750 psig 680°F	B.S. & B 95215-4	PORV
813,814	8" Gate	675198/1 2485 psig 650°F	Crane DR-33463	RCS inlet to RHR
833,834		675198/1 2485 psig 650°F	Crane DR-33473	MOV - Block Valve RHR - Loop Return
850 A,B,C		675198/3 2485 psig 650°F	Crane DR-33473	Safety Inspection Block Valve A,B,C
867 A,B,C	6-C58FC	675268/4 2511 psig 680°F	Edward P-19333	Check Valves
FCV-1115 D,E,F		676044/8 2550 psig 250°F	B.S. & B 95215-7	Seal Water Supply to Reactor Coolant Pump Loop - A,B,C
LCV-1112		676044/3 2200 psig 600°F	B.S. & B 95215-9	Letdown Valve Loop A to Regeneration HX
HCV-1117		676044/3 2200 psig 250°F	B.S. & B 95215-10	Excess Letdown, HX Isolation Valve Main Coolant Outlet
PCV-430H PCV-430C		676044/1 2485 psig 650°F	B.S. & B 95215-1	Pressurizer Spray A, B

TABLE 8-1 VALVES CONSIDERED (cont'd)

Valve Location	Valve ID	W Specification & Sheet	Vendor Drawing No.	Purpose
264 272 280 308 354	2C58EG	675768/4 2511 psig 680°F	Edward C-61462	Check Valves
261 269 277 504 508 513	2T58EE	675268/16 2511 psig 680°F	Edward D-61649	Isolation & Drain Valves
500,501 502,503 506,507 511,512 524,525 528,529 950,952 954,959	3/4 T58ED	675268/16 2511 psig 680°F	Edward D-61577	Isolation Valves
951,953 955,956	3/8 TD 58	676044/1SS 2511 psig 680°F	Edward D-67502	Sampling Valves
202,203 204	-	676044/6 3750 psig 650°F	B.S. & B 95215-20	Letdown Orifice Remote Shut Off Valves

9.0 HEAT EXCHANGERS

This section presents an evaluation of the Excess Letdown and Regenerative Heat Exchangers and the Letdown Orifices at the San Onofre Nuclear Generating Station for an ATWS event consisting of a pressure/temperature transient of 2998 psig at 700°F. The evaluation will be performed by showing that the stress effects due to the ATWS are less than those experienced during the hydrostatic test of the equipment, with account taken for the hydrotest temperature versus the ATWS temperature.

A search of the readily available files concerning this equipment yielded the following information:

Regenerative Heat Exchanger

Manufacturer: Basco, Inc., Buffalo, NY
Westinghouse Purchase Order No. 54-P-41432-B
Westinghouse Specification No. 675233
Outline Drawing: Basco Dwg. B-1-12240, Rev. 6
Shell Side Design Pressure: 2485 psi at 650°F
Shell Side Test Pressure: 4705 psi
Applicable Code: ASME Section VIII, Code Cases 1270 and 1273N
National Board No. 5043
Date Manufactured: 8/65

Excess Letdown Heat Exchanger

Manufacturer: Basco, Inc., Buffalo, NY
Westinghouse Purchase Order No. 54-P-41432-B
Westinghouse Specification No. 675232
Outline Drawing: Basco Dwg. B-1-12243, Rev. 3, and B-1-12308-SCE
Tube Side Design Pressure: 2485 psi at 650°F
Tube Side Test Pressure: 4705 psi
Applicable Code: ASME Section VIII, Code Case 1270N
National Board No. 5049
Date Manufactured: 6/65

Unfortunately, the specific materials of the construction were not located. A review of similar heat exchangers for other older plants (Kewaunee, Point Beach, Trojan) showed that the portions of these heat exchangers that receive water from the reactor coolant loop are typically made of Type 304 stainless steel such as SA-351-CF8, SA-240 T304, SA-182 F304, and SA-213 T304. This includes the shell, tubesheet, and tubes for the regenerative heat exchanger, and the tubes, tubesheet, and tube inlet/outlet for the excess letdown heat exchanger. The orifices are also assumed to be made of similar material. For these materials, Section VIII of the Code gives a ratio of 0.8457 for the allowable stress at 700°F over the allowable stress at 100°F. If we consider that the hydrotest pressure is 1.5 times the design pressure, applied at 100°F the allowable pressure for the ATWS becomes 1.5 x 2485 psi x 0.8457 = 3152 psi. If we use the test pressure given on the heat exchanger drawings of 4705 psi, the allowable ATWS pressure becomes 4705 x 0.8457 = 3979 psi. Both of these numbers are higher than the ATWS condition being evaluated, which is 2998 psi at 700°F. This is sufficient to demonstrate structural integrity under ATWS conditions.

10.0 PRESSURIZER RELIEF TANK

The pressurizer relief tank has been evaluated to determine the effect of pressurizing the tank to bursting. This postulated event would cause the rupture disks, which are designed for over-pressure protection, to function as designed and relieve the pressure before the tank could burst. The results of the evaluation indicate that the pressurizer relief tank will not fail, but if the rupture disks did not function it is possible that the tank could rupture in the area of a nozzle or other discontinuity. In this failure mode the material would separate and vent the internal pressure. The tank material will not fail in a brittle manner and fragmentation will not occur. This evaluation indicates that missiles from the pressurizer relief tank do not result from the postulated over pressurization of this tank.

It therefore may be concluded that the relief tank is unlikely to fail as a result of the ATWS transient, but even if it does, no missiles will be produced and safe plant operation will not be compromised.

11.0 INSTRUMENTATION

Instrumentation which serves as part of the reactor coolant system pressure boundary is limited to resistance temperature detectors (RTDs) and various transmitters. Structural integrity of this instrumentation is demonstrated by a hydrostatic pressure test. Westinghouse identifies requirements for this pressure test in equipment specifications for the instrumentation. This test was performed at a pressure of 3728 psi for San Onofre Unit 1, so therefore the instrumentation will be qualified for the maximum ATWS pressure of 2998 psi.

12.0 SUMMARY AND CONCLUSIONS

The results discussed in this report confirm the capability of San Onofre Unit 1 to withstand a postulated Anticipated Transient Without Scram. These results are consistent with the conclusions of earlier generic submittals on this subject [2,4].

The transient which produces the highest pressure for the San Onofre Unit 1 system was determined to be the loss of normal feedwater ATWS event, which results in a maximum pressure of 2998 psig at 700°F.

The evaluations reported herein have used a consistent methodology, as detailed in Section 2. The ATWS event was treated as a high temperature hydrotest event, with the allowable stress reduced to account for the higher temperature. The maximum ATWS pressure is significantly less than the hydrotest pressure for the system (3728 psig), so the net effect is that the lower pressure counteracts the lower allowable, and the result is that all components have been shown to be acceptable.

13.0 REFERENCES

1. Anticipated Transients Without Scram for Light Water Reactors, NUREG-CR-0460, U.S. Nuclear Regulatory Commission, April 1978.

2. "Anticipated Transients Without Scram for Westinghouse Plants," Westinghouse Electric Corporation Letter NS-TMA-2182, December 1979.

3. "Analysis of Anticipated Transients Without Scram for San Onofre Nuclear Generating Station Unit 1," Westinghouse Electric Corporation, letter SCE-90-553, April 6, 1990 (to be published).

4. "Westinghouse Anticipated Transients Without Trip Analysis," Westinghouse Electric Corporation WCAP 8330, August 1974.

5. Calc: "Qualification of SCE Class 1 Piping and Fittings for ATWS Events," by R. Brice-Nash, dated 4/11/90, SCE-145-14.

6. E-Spec: 676171, Rev. 0, dated 3/12/65, AUXILIARY PIPING SYSTEMS, SAN ONOFRE NUCLEAR GENERATING STATION, UNIT NO. 1.

7. E-Spec: G569866, Rev. 1, dated 8/18/64, MATERIALS SPECIFICATIONS PIPE AND FITTINGS.

8. E-Spec: 675188, Rev. 0, dated 9/6/63, REACTOR COOLANT PIPE, REACTOR COOLANT SYSTEMS, SAN ONOFRE NUCLEAR GENERATING STATION UNIT 1, SCE-140.

9. WCAP-8827, "Seismic Re-analysis and Design of San Onofre Unit No. 1 Modified Reactor Coolant System," Volume 1 & 2, January 1977.

10. EMD Engineering Memorandum No. 4672, "Seismic Analysis Reactor Coolant Pump SV-4M-A1, Southern California Edison Company, San Onofre," March 21, 1975.

11. NRC Docket 50-206, "San Onofre Nuclear Generating Station Unit 1 Seismic Re-evaluation and Modification, April 29, 1977.

12. WTD-SM-75-021, "Time History Seismic Analysis of the Steam Generators for the SCE San Onofre Power Plant," April 1975.

13. WTD-SM-74-007, "Geometry and Weight Calculations for the 27,700 Square Feet Steam Generator," June 1974.

14. EMD Engineering Memorandum No. 4664, SCE RC Pump Seismic Analysis for Faulted Transient Loading.

15. Reactor Vessel Design Report for the San Onofre Nuclear Generating Station Unit 1, Westinghouse Electric Corporation, dated July 1965.

16. PB-151987, Tentative Structural Design Basis for Reactor Pressure Vessels and Directly Associated Components (Pressurized, Water Cooled Systems) U.S. Department of Commerce, 1 December 1958 Revision.

Enclosure 4
Design Description
SONGS 1 ATWS Mitigation System

ATWS MITIGATION SYSTEM

Description of Design

The auxiliary feedwater system (AFWS) consists of two redundant and independent pump trains. Each train is capable of delivering flow to all three steam generators for decay heat removal in the worst design basis event with respect to the AFW flow requirements. Interlocks prevent simultaneous operation of both trains, in the event of a single failure, to maintain flow below the water-hammer limits.

Train B AFWS consists of one motor-driven pump, G-10W. Train B AFWS is the lead train and will actuate on the Train B auxiliary feedwater actuation signal (AFWAS). A motor-driven pump, G-10S, and a turbine-driven pump, G-10, constitute Train A AFWS. Train A AFWS is the lag train and will actuate upon failure of Train B AFWS.

Level transmitters, LT-3400 A, B, and C, and LT-2400 A, B, and C, generate low steam generator level signals for the respective trains when the water level falls below 5% narrow range span in the steam generators. A 2/3 logic from these level transmitters will generate Auxiliary Feedwater Actuation Signal (AFWAS) and actuate the respective AFW trains. LT-3400 A, B, and C, inputs to Train B AFWAS, and Train B AFWS are powered from 120 VAC Vital Bus 5 and 4160 Volt Bus 2C, respectively. LT-2400 A, B, and C, inputs to Train A AFWAS, and Train A AFWS are powered from 120 VAC Bus 3A and 480 Volt Bus 1, respectively.

For the ATWS Mitigation System, only Train B AFWS is credited since it is powered from electrical Train B. The entire Reactor Protection System (RPS) is powered from electrical Train A. Therefore, the SONGS 1 ATWS Mitigation System satisfies the diverse and automatic auxiliary feedwater actuation required by the ATWS rule.

The Train B AFWAS will also actuate the diverse turbine trip (DTT) circuitry which is powered by 125 VDC Bus 2. The DTT will energize a new turbine trip solenoid to drain the turbine auto-stop oil, and trip the turbine within 30 seconds after the receipt of the signal. The ATWS analysis assumes the turbine trips 30 seconds after the receipt of the AFWAS.

Changes in the design are in the safety-related and nonsafety-related interface, and the absence of the operating bypass for the DTT circuitry. In the previous conceptual design, the CCC relays would provide isolation between the safety-related (SR) DTT logic circuitry and nonsafety-related (NSR) turbine trip actuating and annunciating components, and between the P-7 permissive on Train A and the DTT logic circuitry on Train B.

The preliminary design of the ATWS Mitigation System used a SR DTT circuitry. The current design uses the NSR/ATWS DTT circuitry. The isolation function provided by the CCC relays between the DTT circuitry and nonsafety-related annunciating and actuating components is no longer necessary. Foxboro output relay cards provide isolation between the nonsafety-related DTT circuitry from the safety-related power supply.

The P-7 permissive is a part of the RPS and provides an operating bypass below 10% power. The P-7 permissive would provide A bypass of the DTT circuitry below 10% power to prevent spurious actuation where the steam generator level is not stable. Since a turbine trip does not trip the reactor below 10% power, the final design of the DTT circuit does not include the operating bypass capability. It was determined that the use of the bypass did not provide any operational benefits. Hence, P-7 output was deleted in the DTT circuitry to simplify the design and use of the CCC relays was not necessary.

The design of the ATWS Mitigation System is completely independent from the sensor output to final actuating device from the RPS. The ATWS Mitigation System logic diagrams and an elementary are attached.

Attachment
Elementary and Logic Diagrams
SONGS 1 ATWS Mitigation System