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SUBJECT: Responds to NRC 820327 ltr & submits addl info re existing capability for rapid depressurization & decay heat removal w/o power operated relief valves.Info provides justification for full safe power operation.

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April 30, 1982

Director, Office of Nuclear Reactor Regulation
Attention: Mr. Frank Miraglia, Branch Chief
Licensing Branch No. 3
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Gentlemen:

Subject: Docket Nos. 50-361 and 50-362
San Onofre Nuclear Generating Station
Units 2 and 3

The NRC's letter of March 27, 1982, requested SCE to provide additional information relative to the existing capability of San Onofre Units 2 and 3 for rapid depressurization and decay heat removal without Power Operated Relief Valves (PORV's). The letter also indicated that SCE should provide a schedule for responding to the NRC's questions, and that if the responses to all the questions could not be provided at least one month prior to the expected date for operation of San Onofre Unit 2 at power levels above 5%, SCE should provide justification for full power operation of San Onofre Unit 2 in the interim until the issue is resolved.

In response to the March 27, 1982 NRC letter, SCE's letter of April 9, 1982 indicated that, because of the complexity of the NRC questions and the close proximity of the scheduled date for operation of San Onofre Unit 2 above the 5% power level, it was unrealistic to expect responses to the NRC's questions one month prior to the scheduled date for operation above 5% power. SCE's letter further indicated that the Combustion Engineering Owners Group (CEOG) would sponsor preparation of generic responses to the NRC's request and that SCE would provide, on May 1, 1982, a justification to demonstrate that San Onofre Unit 2 can safely operate at full power.

Consistent with the May 1, 1982 commitment, enclosed please find seven (7) copies of the following justification report (NRC Mail Code B028):

ENCLOSURE: A Review of The Depressurization and Decay Heat Removal Capabilities of San Onofre Units 2 and 3.

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The enclosed report provides a justification for safe full power operation of San Onofre Unit 2 based on the following considerations which are discussed in detail in the report:

1. The San Onofre Units 2 and 3 NSSS's are coupled with a highly reliable, safety grade Auxiliary Feedwater System (AFWS). The AFWS design for San Onofre Units 2 and 3 exhibits a higher level of reliability than most other AFWS designs.

2. San Onofre Units 2 and 3 are capable of achieving cold shutdown conditions using only safety grade systems, even without offsite power and with an additional single failure.

3. The San Onofre Units 2 and 3 steam generator design includes many features which will enhance tube integrity, minimizing concerns associated with operating reactors. Additionally, careful attention to the plant water chemistry program will ensure that the magnitude of the impurity ingress into the steam generators is maintained at a low level. Because of the steam generator water chemistry program and design features which minimize steam generator tube corrosion and stress, SCE considers that steam generator tube degradation should not be a concern during the period the NRC questions are being addressed.

4. Even if all auxiliary feedwater supply were somehow lost, heat removal could still be achieved by depressurizing the steam generators to allow the use of low head pumps (e.g., firewater or condensate pumps).

5. Review of probalistic analyses conducted by the NRC do not show any justification for the addition of Reactor Coolant System (RCS) valves for decay heat removal purposes.

SCE considers that the enclosed report adequately responds to the NRC's request for information to justify full power operation of San Onofre Unit 2 without PORV's in the interim, until the issue of the adequacy of rapid depressurization and decay heat removal capability for the existing San Onofre Units 2 and 3 design is resolved.

Relative to the preparation of responses to the fourteen (14) questions forwarded by the NRC March 27, 1982 letter, the CEOG is currently identifying the scope of work and schedule for responding to generic questions. Concurrently, SCE is evaluating the questions to identify plant specific concerns which will be addressed separately from the CEOG effort. SCE will advise the NRC as soon as the scope of work and schedule for preparation of responses are clearly defined.

If you have any questions or comments, please let me know.

Very truly yours,

R P Baskin

A REVIEW OF THE DEPRESSURIZATION AND DECAY HEAT REMOVAL CAPABILITIES OF SAN ONOFRE UNITS 2&3

1.0 INTRODUCTION

The NRC has requested that Southern California Edison Co. (SCE) provide an evaluation of the rapid depressurization and decay heat removal capabilities of the San Onofre Units 2 and 3 design. This evaluation is to be similar to that requested of Combustion Engineering (CE) on the CESSAR docket, incorporating the answers to questions asked on that docket. SCE is participating with the CE Owners Group (CEOG) for the development of responses to the NRC's questions. SCE and the CEOG are currently identifying the scope of work and schedule for responding to the NRC's questions. SCE will advise the NRC as soon as the scope of work and schedule for preparation of responses are clearly defined. Additionally, the NRC requested that SCE provide a justification for safe operation of the plant at full power during the period of this evaluation. This report provides justification for safe full power operation of San Onofre Units 2 and 3 based on the following considerations listed below, which are amplified in the report:

1. The San Onofre Units 2 and 3 NSSSs are coupled with a highly reliable, safety grade Auxiliary Feedwater System (AFWS). The AFWS design for San Onofre Units 2 and 3 exhibits a higher level of reliability than most other AFWS designs.

2. San Onofre Units 2 and 3 are capable of achieving cold shutdown conditions using only safety grade systems, even without offsite power and with an additional single failure.

3. The San Onofre Units 2 and 3 steam generator design includes many features which will enhance tube integrity, minimizing concerns associated with operating reactors. Additionally, careful attention to the plant water chemistry program will ensure that the magnitude of the impurity ingress into the steam generators is maintained at a low level.

Because of the steam generator water chemistry program and design features which minimize steam generator tube corrosion and stress, SCE considers that steam generator tube degradation should not be a concern during the period the NRC questions are being addressed.

4. Even if all auxiliary feedwater supply were somehow lost, heat removal could still be achieved by depressurizing the steam generators to allow the use of low head pumps (e.g., firewater or condensate pumps).

5. Review of probalistic analyses conducted by the NRC do not show any justification for the addition of Reactor Coolant System (RCS) valves for decay heat removal purposes.

2.0 BACKGROUND

The early CE NSSS designs used Power Operated Relief Valves (PORVs) as non-safety grade equipment to limit overpressure transients to pressures below the ASME Code safety valve setpoint. This function was intended to reduce challenges to the safety valves, thereby minimizing weepage and avoiding potential leakage following actuation. The PORVs were not intended to prevent a high pressure reactor trip, but rather, were to be used in conjunction with the trip to mitigate the pressure transient.

As each of the early plants became operational, the effectiveness of the pressurizer spray system to limit pressure transients was demonstrated. Consequently, CE was unable to substantiate any advantages to opening PORVs during transients to protect the safety valves from leakage. PORVs were also considered to be counterproductive in light of the PORV leakage problems that had been experienced. Furthermore, system analysis has demonstrated the pressure overshoot above the high pressure trip to be so minimal that, when PORV operation was not credited, the safety valves were still not challenged.

Accordingly, the PORV function during power operation was not considered necessary, and was eliminated from subsequent CE designs.

Recently, a contigency method of core cooling employing once-through flow in the RCS has been advanced as an alternate decay heat removal system. This method would use PORVs in conjunction with the High Pressure Safety Injection (HPSI) pumps and has been referred to as "feed and bleed." In this regard, the Advisory Committee on Reactor Safeguards (ACRS), following its review of CE's System 80, (which is similar to San Onofre Units 2 and 3 in this regard) stated:

> "In recent years, the availability of reliable shutdown heat removal capability for a wide range of transients has been recognized to be of great improtance to safety. The System 80 design does not include capability for rapid, direct depressurization of the primary system or for any method of heat removal immediately after shutdown which does not require use of the steam generators. In the present design, the steam genertors must be operated for heat removal after shutdown when the primary system is at high pressure and temperature. This places extra importance on the reliability of the auxiliary feedwater system used in connection with System 80 steam generators and extra requirements on the integrity of the steam generators. The ACRS believes that special attention should be given to these matters in connection with any plant employing the System 80 design. The Committee also believes that it may be useful to give consideration to the potential for adding valves of a size to facilitate rapid depressurization of the System 80 primary coolant system to allow more direct methods of decay heat removal. The Committee wishes to review this matter further with the cooperation of Combustion Engineering and the NRC Staff."

In meetings with the ACRS and NRC Staff, CE has presented its position and the bases for its design. The NRC has raised a series of concerns regarding this issue and provided a list of questions to CE and applicant utilities. In recognition of the scope of these questions the NRC has requested justification for opertion during the period of time the questions are being addressed. The ACRS has agreed with this approach stating that:

"....while this evaluation should be conducted expeditiously its resolution should not now be a condition for operation of System 80 plants at full power or of plants having similar features."

During a recent meeting of the Combustion Engineering Owners Group (CEOG), it was agreed that the CEOG would sponsor preparation of generic responses for affected CE Utilities. This submittal provides justification for full power operation of San Onofre Units 2 and 3 during the period of time that these questions are being addressed.

3.0 AUXILIARY FEEDWATER SYSTEM RELIABILITY

The San Onofre Units 2 and 3 NSSS design is coupled with a safety grade Auxiliary Feedwater System which has been subject to extensive development by SCE, CE, and Bechtel. This system in conjunction with the safety grade atmospheric dump valves provides an assured method of RCS heat removal. Detailed design reviews were conducted in 1977 and again in 1981 (post TMI) and in both cases significant modifications and upgrades were implemented. In 1977 cavitating venturies and low flow bypass valves were added to enhance AFW flow characteristics. Following the post TMI review a third full capacity pump was added to improve the system reliability and to fully meet high energy line break criteria. The AFW system, which is documented in the San Onofre Units 2 and 3 FSAR, is a three train system with one train independent of ac power. It is seismic category 1, electrical class 1E and designed to ASME code class 2 and 3. It has the pump driver diversity and the independence and separation for pipe break consideration necessary to meet Branch Technical Position ASB 10-1. The AFWS design for San Onofre Units 2 and 3 exhibits a higher level of reliability than most other AFWS designs. In its Safety Evaluation Report the NRC stated that:

"We conclude that the San Onofre Units 2 and 3 AFWS will have a high degree of reliability..."

Although no quantitive requirement for expected system availability was explicity imposed, the San Onofre 2 and 3 AFWS design reflects the high reliability needed to meet the current SRP criteria of unavailabilities in the range of 10^{-4} to 10^{-5} per demand.

4.0 CAPABILITY TO ACHIEVE COLD SHUTDOWN

There are numerous systems, both within the NSSS design and BOP design, for San Onofre Units 2 and 3 available to perform the various functions necessary to bring the plant to a cold shutdown condition. As a group, these systems provide the operator with the flexibility necessary to cool down and depressurize the plant in a variety of possible situations. The design meets Branch Technical Position RSB 5-1 as documented in responses to NRC questions 212.139, 212.157 and 212.164 in the San Onofre Units 2 and 3 FSAR. Some of the more significant features of the San Onofre Units 2 and 3 design related to shutdown, cooldown, and depressurization capabilities are discussed below.

Normal Shutdown:

Under the vast majority of situations, the same systems used for power generation will be employed for plant cooldown. In these cases primary coolant is circulated through the RCS using the reacor coolant pumps. Steam is drawn from the steam generators, bypasses the turbine and is rejected to the main condenser. The main feedwater and condensate systems are used to return the condenser inventory to the steam generators. RCS heat removal is maintained with the steam generators. RCS pressure is maintained with the pressurizer, using the normal heater and spray control systems.

Shutdown with Heat Rejection to Atmosphere:

In the event that the main condenser or associated systems are unavailable, steam may be rejected directly to atmosphere. Either of two safety grade steam generator atmospheric dump valves located upstream of the MSIVs may be operated manually to bleed steam. Makeup water to the steam generator is supplied from either the Main Feedwater System or the safety grade AFWS. This system provides sufficient inventory to allow for plant cooldown (i.e., sensible heat removal) and decay heat removal for a period of time in excess of 24 hours. Additional makeup from other site sources allows for extenaed operations.

Natural Circulation:

Central to the accomplishment of the basic safety function of Core Heat Removal is the ability to transport reactor coolant to a heat sink where RCS Heat Removal can be accomplished. Reactor coolant pump forced circulation and heat transfer to the steam generators is the preferred mode of operation for residual heat removal whenever plant temperatures and pressures are above the shutdown cooling system (SDCS) entry conditions. Subcooled natural circulation provides an effective alternate means for controlled core cooling, using the steam generators, for extended periods of time if the reactor coolant pumps are unavailable. Two-phase natural circulation and reflux cooling will also occur to provide adequate core cooling following transients which result in loss of RCS inventory and/or subcooling.

Component elevations of San Onofre Units 2 and 3 are such that satisfactory natural circulation for decay heat removal is obtained as a result of density differences between the bottom of the core and the top of the steam generator tube sheet, an elevation head of approximately 25 feet. An additional small contribution to natural circulation flow rate is the density difference obtained as the coolant passes through the steam generator U-tubes. Additionally, several systems design features have been incorporated to assure the maintenance of natural circulation flow. A redundant pressurizer heater capacity of 150 KW from each diesel genertor is available to maintain system subcooling. A reactor coolant gas vent system is provided to allow the purging of noncondensible gases should they form. Additionally, natural circulation plant performance will be extensively tested during the start-up period of San Onofre Unit 2.

When in natural circulation, the main pressurizer spray system is unavailable. The safety grade auxiliary spray from the charging system provides for system depressurization under these conditions. This system has been modified to provide an independent manual bypass. Thermal shock considerations are addressed by the use of a thermal sleeve in the spray nozzle. CE recommends use of the auxiliary spray system for primary depressurization whenever the main pressurizer spray system is unavailable.

In summary, the San Onofre Units 2 and 3 design meets Branch Technical Position RSB 5-1, "Design Requirements of the Residual Heat Removal System" as described above. San Onofre Units 2 and 3 can be brought to SDCS initiation in less than 36 hours using only seismic category 1 equipment, assuming the most limiting single failure, and with only onsite or only offsite power available.

5.0 STEAM GENERATOR INTEGRITY

The 3410 MWt steam generators are of an improved design selected to mitigate or resolve operating problems which have been experienced with U-tube steam genertors of the recirculation type. The general arrangement is similar to currently operaing 2570 MWt CE systems including a number of design improvements and retained features to assure improved operational reliability and maintenance of integrity for decay heat removal after reactor shutdown.

The 3410 MWt steam generator is of the vertical U-tube, natural recirculation, non economizer type and is somewhat larger than the earlier 2570 MWt steam generator and contains approximately 9,350 tubes instead of 8,400 tubes.

The design as it affects secondary side hydraulics has been improved to remove areas of possible localized dryout. This has been accomplished by a number of modifications in the tube bend region:

1. The vertical tube spacer strips have been separated from the diagonal "bat wing" tube supports.

2. The "bat wing" supports have been lowered to avoid intersecting the tube bends.

3. The tube supports in the small radius bend region have been located below the bends.

4. The vertical tube spacer strips are now provied with large "punchouts" to enhance cross flow freedom.

5. The former drilled upper tube support plates have been replaced with partial "eggcrate" type supports.

Thus all tube supports are of the "eggcrate" or lattice type to promote freedom of vertical as well as cross flow.

The elimination of the drilled upper tube support plates will mitigate the denting problems previously experienced in this region.

The Inconel 600 mill annealed tubing is specified, controlled and tested in a manner to preclude sensitivitity to stress corrosion cracking or intergranular attack. Subsequent CE shop tube fabrication practices utilize carefully controlled and proven techniques to minimize residual tube stress, a contributor to stress corrosion cracking. These include:

1. The bending techniques used are selected to minimize residual tube stress. CE has historically used a relatively large tube bending radius for the inner tube rows.

2. CE uses the explosive technique for placing the tube in contact with the tubesheet for the full tubesheet thickness. This elimiates the tube-to-tubesheet crevice which has caused corrosion problems in this region, such as stress cracking and intergranular attack.

The steam generator design allows for sludge lancing to periodically remove accumulations of solids from the upper tubesheet face. These sludge accumulations have been the site of tube pitting type attack.

CE utilizes a mechanical joint between the primary head divider plate and its juncture with the tubesheet and primary head. This eliminates the possibility of the differntial growth and deflection between these members causing tubesheet clad separation and tube damage which has occurred in non CE units.

The 3410 MWt design utilizes large top discharge elbows for the main/auxiliary feedwater inlet sparger. In addition the drain time of this sparger ring has been increased by a sealing device located between the sparger and the feedwater inlet nozzle. Thus water hammer potential with possible feedwater line damage is reduced.

The integrity of the steam generator tubing is also protected through the use of strict controls on the steam generator water chemistry. The chemical environment of the steam generator secondary side is monitored and controlled during all phases of plant operations including power operation, startup, shutdown, and maintenance outages.

Steam Generator chemistry is maintained through a combination of control of impurities delivered to the steam generator, monitoring and controlling the chemical environment within the steam genertor, and removal of any materials which may be introduced. Through feedtrain features and procedures, including a high integrity condenser, startup recirculation, and chemical addition, the magnitude of impurity ingress into the steam generator is maintained at a low level. A chemistry control program is employed to assure that secondary water chemistry is maintained within appropriate control bounds during operation and that timely corrective actions are taken in the event abnormal chemistry occurs. An all volatile treatment water chemistry is utilized for the secondary systems. This method of secondary chemistry control precludes tube corrosion and related problems due to the chemical additives, and it minimizes the amount of sludge deposited within the steam generator. Routine corrective actions for abnormal chemistry include increasing the steam generator blowdown rate, adjustment of chemical addition rates, and more extensive monitoring of steam generator chemistry. For severe upset conditions, power reduction and/or plant shutdown is specified. Continuous sampling of and chemical addition to the steam generator monitors the effectiveness of feedtrain impurity controls and maintains a chemical environment condusive to low corrosion rates within the steam generator. Finally, steam generator blowdown, supplemented by fill and drain when required, serves to remove those impurities which are introduced. By minimizing contaminent ingress, monitoring system performance, and taking corrective action when necessary, chemistry related challenges to the integrity of the steam generator tubes are minimized.

During accident reponse conditions, water supplied to the steam generator by the Auxiliary Feedwater System originates in the condensate storage tank. This makeup quality water is chemically treated and its use will not challenge the steam generator tube integrity. In the unlikely event that water must be supplied from alternate sources during the accident (fire water, etc.) it is not anticipated that even this impure water will cause tube failure in the time frame of the accident and subsequent plant cooldown.

In summary it is considered that the design, material and manufacturing features discussed above, along with appropriate chemistry control, will assure improved steam generator tube integrity. SCE further considers that steam generator tube degradation should not be a concern during the period the NRC questions are being addressed.

6.0 CONTINGENCY DECAY HEAT REMOVAL (DHR)

The San Onofre Units 2 and 3 design meets current licensing criteria with regard to DHR capabilities. The consideration of additional RCS valves for DHR essentially addressed contingency (or "last resort") capabilities that go beyond existing design bases. In this regard it is significant to note that a potential already exists for contingency heat removal by depressurizing the steam generators.

The potential mode of plant operation considered is as follows: Following reactor trip and the very unlikely event of a total loss of all feedwater, the plant could be brought to hot standby using either the secondary safety valves or the atmospheric dump valves. The safety grade steam generator atmospheric dump valves then provide the contingency capability to blowdown and depressurize the steam generator secondary system. At the reduced steam generator pressure a low head pump (e.g., firewater or condensate pump) could be aligned to deliver feed to the steam generator. Then, with sufficient feedwater and steam flow, continuous decay heat removal could be established at those "off design" conditions.

There appear to be several advantages to steam generator depressurization in preference to primary feed and bleed. These are:

1. <u>The reactor coolant pressure boundary is maintained intact.</u>

   Therefore the potential radiological release to the containment and possibly to the environment is avoided. Any necessary containment entry for repairs would not be impeded. Additionally the large clean-up cost that would be associated with the use of primary feed and bleed is avoided.

2. <u>There is time available for operator action.</u>

   Delivery of secondary makeup to a depressurized steam generator can be accomplished any time prior to core uncovery, which is estimated to be approximately 90 minutes, to ensure adequate core cooling.

3. <u>Equipment involved is accessible.</u>

   The atmospheric dump valves and various low head pumps are located outside containment where access for maintenance and repair is possible. PORVs on the other hand would be inside containment and virtually inaccessible.

4. <u>Procedures are consistent with normal DHR procedures.</u>

Normal procedural efforts focus upon restoration of feedwater. Initiation of primary feed and bleed would represent a dramatic departure from this strategy.

The final reason noted above is worthy of elaboration in that it was strongly supported by plant operators during procedure work shops conducted at CE. Plant operators feel that it is highly preferable to continue operation with the steam generators performing the function of RCS Heat Removal, while the functions of RCS Inventory and Pressure Control are being controlled separately. With the initiation of RCS feed and bleed all three safety functions would now rely on a single process with no degree of independent control. The extreme difficulity in dealing with the competing demands of RCS Heat Removal, Pressure and Inventory Control by regulating a single process has been clearly demonstrated at TMI-2 and Ginna.

7.0 PROBABILISTIC JUSTIFICATION (REVIEW OF DRAFT PRA)

A draft PRA provided to CE by the NRC Division of Risk Analysis (DRA) attempted to demonstrate that the CE plants which lack a capability for core cooling via feed and bleed operation will not meet the NRC's proposed plant performance guideline. This guideline is that "the likelihood of a nuclear reactor accident that results in a large-scale core melt should normally be less than one in 10,000 per year of reactor operation." Additionally, the DRA study made a case for incorporating feed and bleed capability to partially alleviate the perceived problem, and presented analysis to show that such a change is cost beneficial to the utilities.

Review of the draft PRA (which has since been characterized as "overstated" by the author) indicates that the recommendations are not well supported by the analyses. The following comments are offered:

General Comments

1. The NRC proposed safety goal was developed in the light of PRA analyses which have all been done assuming some nominal plant age, that is, an age for which the usual assumptions inherent in reliability analyses apply. The DRA study uses the same safety goal to apply to very early plant operation that can be characterized as the wear-in period rather than applying the goal to average plant conditions. This appears to be a misapplication of the safety goal.

2. The DRA study includes treatment of uncertainty and shows that, given huge uncertainty spans (three orders of magnitude), the upper bound estimate may somewhat exceed the plant performance guideline. This approach is in conflict with the NUREG-0880 recommendation of the Staff regarding treatement of uncertainties. NUREG-0880 recommends that probabilistic risk assessments be performed during the trial period on the basis of "realistic assumptions and best estimate analyses."

3. The NRC's "Proposed Policy Statement on Safety Goals for Nuclear Power Plants" states, under the heading of "Implementation," that the proposed numerical cost/benefit guidelines may be used by the NRC staff during the trial period, and that benefits should be measured in radiological risk. Costs should be annualized over the remaining plant life.

   However, the cost/benefit analysis contained in the DRA study does not agree in form or content with the above policy. Most importantly, consideration was not limited to radiological risk. Since no radiological consequences were predicted for the events considered, the only benefit identified by the DRA is a reduction in the utility's economic risk. Cost/Benefit based on utility economic risk is outside of the intended scope of the guidelines and should not be the basis for developing NRC requirements.

   Cost/benefit based on utility economic risk is clearly a serious misapplication of the safety goal.

Specific Comments

1. The draft PRA discussed three potential accident sequences for which plants lacking feed and bleed capability may not meet the safety goal. These are listed below, together with reasons why it is believed that they are not applicable to San Onofre Units 2 and 3.

   Although, "back-of-the-envelope" calculations require simplified and conservative assumptions, such assumptions resulted in erroneous conclusions reached by the Staff in their analysis. Specifically, it was assumed that only one diesel generator is capable of energizing the safety related motor-driven AFWS train and that offsite power is required for the other motor-driven AFWS train. In fact, the current San Onofre Units 2 and 3 AFWS design does not have this configuration.

   a. Total Loss of Feedwater

      The conclusion of the write-up on total loss of feedwater is that "even at maturity this core melt sequence frequency may be higher than 10^{-4}/year." This conclusion is the direct result of the enormous uncertainty band chosen by the analyst. There are three orders of magnitude in the uncertainty of the core melt frequency due to loss of main feedwater (2.6×10^{-4} - 3.9×10^{-7}). By arbitrarily increasing the uncertainty bounds, one can show that any system or event may not meet any goal. As discussed in the general comments above, it is recommended that best estimate calculations be used to demonstrate compliance with the NRC's proposed safety goal. Additionally, the calculation should be based on plant designs that are appropriate to both CE plants and the specific AFWS designs.

   b. Loss of Offsite Power

      The results of this analysis indicate that plants which lack feed and bleed capability are acceptable as long as both motor driven AFW pumps can be powered by diesel generators. As this is the case for

San Onofre Units 2 and 3, the frequency of core melt resulting from loss of offsite power can be considered to be well below the proposed NRC safety goal.

c. Very Small (S_2 LOCA)

This section suggests that all PWRs may suffer from a common problem: that the frequency of core melt due to small break LOCA may exceed the NRC's proposed goal of 10^{-4}/year.

The scenario posed is a S_2 LOCA followed by failure of the Safety Injection System. The combined frequency is estimated by the Division of Risk Assessment at 1.5×10^{-4}/year. There is a short discussion (on page 7) of High Pressure Safety Injection (HPSI) reliability (e.g., 8.6×10^{-3} for Surry, 10^{-3} for Oconee, 10^{-3} for "Most PRAs".) However, this does not reflect the reliability of the San Onofre Units 2 and 3 HPSI system. The design is simpler and more reliable than those evaluated for Surry, Oconee, et al. The San Onofre Units 2 and 3 HPSI design is a single purpose, multi-train system that does not have the potential for the failure modes that have tended to dominate the unreliability estimates of other HPSI systems. Due to these differences alone we believe that the NRC's estimate of 5×10^{-3} per demand is much too high. A best estimate of core melt frequency due to S_2 LOCA at the San Onofre Units 2 and 3 is much less than 10^{-4}. It seems inappropriate to draw conclusions on CE designed systems from the results of analyses on non CE plants.

2. The analysis presented by the NRC is for loss of residual heat removal leading to core melt. The correct conditional failure probabilities should be used for this analysis. Most AFWS reliability analyses have been performed to the requirements specified in NUREG-0635. This document specifies 20 minutes for generator boil dry time as a failure criterion. This criterion is too restrictive for analysis of rare

occurrences such as core melt and its associated risk. To ensure adequate core cooling, it is estimated that the AFWS need only be started within approximatley 90 minutes after total loss of feedwater. This longer time interval permits manual actions, repairs, and restorations of vital support systems and would produce much higher reliabilities than those predicted by NUREG-0635 analysis.

3. The failure probability of the diesel should also be reevaluated. The normal failure criterion for the diesels is that they should be started and loaded in 10 seconds. This criterion might be appropriate for a large break LOCA but is inapproproate for analysis of residual heat removal systems. The 90 minute criterion mentioned above is more correct. This criterion would again produce a much higher diesel reliability than that used by NRC in their analysis.

4. The use of error bands in the NRC analysis seems unconventional and inappropriate. The meaning of the error bands or how they were generated or combined is not clear. Their appropriateness to the analysis and the safety goal is also questionable. Most analyses of core melt risk have been best estimate calculations. Although the methodology of compliance with the proposed safety goal has not been defined, it should be based on best estimate calculations. The use of undefined error bands and their comparison with the proposed safety goal is not appropriate.

5. The cost benefit analysis prepared by the Division of Risk Analysis appears to be seriously flawed. As discussed in the general comments above, the scope of the analysis is not in keeping with the NRC Proposed Policy Statement. Cost/benefit based on utility economic risk, rather than safety risk to the public, is a serious misapplication of the safety goal. Additionally, the results are misleading in concluding that incorporation of feed and bleed capability would be cost-beneficial. Specifically:

   a. Inappropriate average core melt frequencies are used in the analysis.

b. Costs associated with delayed start-up or plant unavailability due to retrofit are neglected. These could amount to $150 million per plant.

c. The effects of interest payments are neglected in the analysis.

d. Costs associated with maintenance, training, procedures, and routine plant unavailability due to incorporation of feed and bleed capability are not considered in the analysis.

Based on the above comments it is considered that if a corrected analysis was to be performed there would be no apparent justification for plant modification.

8.0 CONCLUSIONS

As requested, a review of the San Onofre Units 2 and 3 design has been completed and the following determined:

1. The San Onofre Units 2 and 3 NSSSs are coupled with a highly reliable emergency feedwater system, with an unavailability in the range of 10^{-4} to 10^{-5} per demand.

2. San Onofre Units 2 and 3 are capable of achieving cold shutdown conditions using only safety grade systems, even without offsite power and with an added single failure.

3. The San Onofre Units 2 and 3 steam generator water chemistry program and design features will minimize steam generator tube corrosion and stress. Additionally, SCE considers that steam generator tube degradation should not be a concern during the period the NRC questions are being addressed.

4. Even if all auxiliary feedwater supply were somehow lost, the potential exists for DHR by depressurizing the steam generators to allow use of low head pumps.

5. Contrary to the draft probability analysis developed by DRA, there is no reason to believe that installing PORV's will result in a significant improvement in safety. The added costs do not appear to be justified.

Based upon the considerations listed above, it is concluded that the current San Onofre Units 2 and 3 design, provides adequate protection for the health and safety of the public and full power operation of San Onofre Units 2 and 3 are fully justified while responses are being prepared to the NRC request for additional information associated with the rapid depressurization and decay heat removal capabilities for San Onofre Units 2 and 3.

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