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POWER SYSTEMS

Docket No.: STN-50-470F

May 26, 1982
LD-82-058

Mr. Darrell G. Eisenhut, Director
Division of Licensing
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Subject: Rapid Depressurization and Decay Heat Removal Capability

References: (A) Letter A. E. Scherer to R. L. Tedesco, LD-82-046, dated April 19, 1982

(B) Letter A. E. Scherer to D. G. Eisenhut, LD-82-029, dated March 4, 1982

(C) Letter R. L. Tedesco to A. E. Scherer, dated March 26, 1982

Dear Mr. Eisenhut:

Combustion Engineering (C-E) indicated in Reference (A) its intention to provide a submittal to justify continued licensing of CESSAR-F while the Staff completed their consideration of the potential for adding valves of a size to facilitate a more rapid depressurization of the System 80 reactor coolant system.

In Reference (B), C-E provided the results of a System 80 design review which concluded that its rapid depressurization and decay heat removal capabilities were sufficient. The Staff subsequently requested additional information in Reference (C). Since this information cannot be provided prior to the issuance of the Final Design Approval (FDA) as stated in Reference (A), C-E agreed to provide a submittal justifying continued licensing of CESSAR-F, and applications referencing CESSAR-F, until this issue is resolved.

The enclosed submittal is similar to that provided by Southern California Edison on the San Onofre Units 2 and 3 which justifies full power operation while the NRC continues consideration of this issue. C-E expects that receipt of this submittal by the NRC will enable the Staff to issue the CESSAR FDA prior to final resolution of the depressurization issue, as recommended in the ACRS letter of April 5, 1982. C-E will continue to prepare responses to NRC questions, as agreed in Reference (A).

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If I can be of any additional assistance in this matter, please contact me or Mr. G. A. Davis of my staff at (203)688-1911, Extension 2803.

Very truly yours,

COMBUSTION ENGINEERING, INC.

G A Davis for

A. E. Scherer
Director
Nuclear Licensing

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Enclosure

A REVIEW OF THE
DEPRESSURIZATION AND DECAY HEAT REMOVAL
CAPABILITIES OF THE COMBUSTION ENGINEERING SYSTEM 80 UNITS

1.0 INTRODUCTION

The NRC has requested that Combustion Engineering (C-E) provide an evaluation of the rapid depressurization and decay heat removal capabilities of the System 80 design. This evaluation is similar to that provided by Southern California Edison (SCE) on the San Onofre Units 2 and 3 dockets. However, the concern was originally raised on the System 80 (CESSAR) docket. C-E is working with the C-E Owners Group (CEOG) for the development of responses to the NRC's questions. The CEOG is currently identifying the scope of work and schedule for responding to the NRC's questions. C-E 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 C-E provide a justification for continued licensing of the System 80 design during the period of this evaluation. This report provides justification for continued licensing of the System 80 design based on the following considerations listed below, which are amplified in the report:

1. The System 80 NSSSs will be coupled with a highly reliable, safety grade Auxiliary Feedwater System (AFWS). The high level of reliability will be assured by adherence to an AFWS reliability interface requirement.

2. The System 80 units are capable of achieving cold shutdown conditions using only safety grade systems, even without offsite power and with an additional single failure.

3. The steam generator design of the System 80 units includes many features which will enhance tube integrity, minimizing concerns associated with operating reactors. Additionally, careful attention to C-E's guidelines for 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, C-E 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 probabilistic 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 C-E 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, C-E 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 C-E designs.

Recently, a contingency 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 System 80 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 importance 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 generators 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, C-E 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 C-E and applicant utilities. In recognition of the scope of these questions the NRC has requested justification for continued licensing 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 C-E Utilities.

3.0 AUXILIARY FEEDWATER SYSTEM RELIABILITY

The C-E Standard System 80 design contains specific BOP interface requirements for an Engineered Safety Features grade Emergency Feedwater System. Although there is currently no quantitative requirement for an expected system unavailability, the deterministic interface requirements reflect the highly reliable system needed to meet unavailabilities in the range of 10^{-4} to 10^{-5} per demand. C-E has worked closely with the System 80 owners in the design of the AFWS and feels confident that the BOP designs will have the high reliability of the AFWS and will add the following interface requirement to CESSAR.

> "The Emergency Feedwater System (EFWS) shall have an unavailability in the range 10^{-4} to 10^{-5} per demand based on an analysis using methods and data presented in NUREG-0611 and NUREG-0635. Compensating factors such as other methods of accomplishing safety functions of the EFWS or other reliable methods for cooling the reactor core during abnormal conditions may be considered to justify a larger unavailability of the EFWS". (1)

4.0 CAPABILITY TO ACHIEVE COLD SHUTDOWN

There are numerous systems, both within the standard NSSS design and BOP design, 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 fully meets Branch Technical Position RSB 5-1. Some of the more significant features of the C-E System 80 design related to shutdown, cooldown, and depressurization capabilities are discussed below.

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(1) This reliability goal is consistent with the acceptance criteria of Standard Review Plan 10.4.9 which is referenced in item II.E.1.1 of NUREG-0737.

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 reactor 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. Any of four safety grade steam generator atmospheric dump valves located upstream of the MSIVs may be operated manually to bleed steam. Makeup water to the steam generators is supplied from the safety grade EFWS. This system provides a safety grade capacity of at least 300,000 gallons of water. This is sufficient inventory to allow for a plant cooldown (i.e., sensible heat removal) and decay heat removal for a period of time in excess of 15 hours. Additional makeup from other site sources allows for extended 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 (SCS) 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 the System 80 plants 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 generator is available to maintain system subcooling. A reactor coolant gas vent system is provided to allow the purging of noncondensible gases should they form. As was done for all other C-E plants, the Standard System 80 natural circulation performance will be tested during the plant startup.

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. Thermal shock considerations are addressed by the use of a thermal sleeve in the spray nozzle. C-E recommends use of the auxiliary spray system for primary depressurization whenever the main pressurizer spray system is unavailable.

In summary, the Standard System 80 design meets Branch Technical Position RSB 5-1, "Design Requirements of the Residual Heat Removal System" as described above. The plants 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 System 80 steam generators are of an improved design selected to mitigate or resolve operating problems which have been experienced with U-tube steam generators of the recirculation type.

Particular attention has been given features which affect tube integrity. Operating experience of competitor and earlier C-E steam generator designs

along with knowledge gained through C-E and industry research and development programs has been carefully incorporated into the System 80 steam generator component design and interfacing system design requirements. These features, combined with careful attention to operating limits and a comprehensive preventative maintenance program are expected to provide a highly reliable steam generation system for the plant lifetime.

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 provided 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 System 80 design of the tube bundle support spacing provides tube bundle rigidity sufficient to prevent vibration fretting and tubing failures following

postulated loss of primary or secondary fluid accidents concurrent with the safe shutdown earthquake without adding significant flow resistance or disturbance. All tube supports (and flow distribution plates) are fabricated from ferritic stainless steel.

The steam outlet nozzles have been provided with an integral venturi type flow restrictor which, in the event of a main steam line break, will greatly reduce secondary side discharge rates. This in turn dampens flow loading on steam generator tubing and internals.

The Inconnel 600 mill annealed tubing is specified, controlled and tested in a manner to preclude sensitivity to stress corrosion cracking or intergranular attack. Subsequent C-E 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. C-E has historically used a relatively large tube bending radius for the inner tube rows.

2. C-E uses the explosive technique for placing the tube in contact with the tubesheet for the full tubesheet thickness. This eliminates the tube-to-tubesheet crevice which has caused corrosion problems in this region.

The System 80 steam generator has high capacity blowdown capability which permits the periodic on-line removal of solids which may accumulate on the tubesheet. As discussed below, flow distribution baffles prevent the dropout of particulates within the tube bundle, but encourage dropout in the region between the inner row of U-tubes and the center of the tubesheet. The high capacity blowdown ducting is located here. When connected to external piping and tankage, this ducting provides the capability to increase the blowdown flow for short periods of time to flow rates approaching five percent MSR (Main Steam Rate) while at full power or nine percent MSR while at hot standby. Flow

rates of this magnitude produce sudden local velocity increases adjacent to the tubesheet of sufficient magnitude to re-entrain particulates dropped from the recirculating flow. Periodic use of the blowdown system prevents the accumulation of corrosion products on the tubesheet.

The System 80 steam generator design includes a carefully optimized axial flow economizer which capitalizes on the heat transfer economy of preheating without increasing the propensity for tube vibration over earlier C-E downcomer-feed steam generators. Local secondary side cross flow feedwater velocities are maintained at a fraction of the recirculating crossflow velocities existing in downcomer-feed steam generators operating for numerous years without tube vibration problems. This is accomplished by careful distribution of the feedwater inlet to the tube bundle around the entire periphery of the cold leg side. Additional assurance against tubing vibration is provided by locating the feedwater inlet to the tube bundle immediately above the tube sheet where the tubing is stiffer than at any other location in the steam generator. After flowing into the tube bundle the feedwater is distributed uniformly across the cold side of the tube bundle by the flow distribution baffle described below and permitted to rise vertically parallel to the axis of tubing. Uniform distribution again provides local axial velocities which are a fraction of those existing in operating units resulting in vibratory driving forces well below those known to be acceptable. Design optimization in System 80 has therefore provided a very large margin of conservatism against damaging tube vibration while resulting in only a neglible reduction in maximum theoretical heat transfer effeciency. The feedwater inlet distribution box is provided with top venting to prevent possible steam accumulation during low load operation and resulting water hammer pulses during switch over from auxiliary (downcomer) feedwater to the main feedwater system.

Flow distribution plates are arranged above the tube sheet with the aid of a highly sophisticated, multi-node, three-dimensional, two-phase, non-homogenous thermal hydraulic simulation, in a manner which insures a uniform distribution of flow across the tube bundle. The baffles also ensure that local horizontal velocities near the tube sheet will permit dropout of boiler water particulates only in the region of the blow down (crud removal) system.

C-E 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 differential growth and deflection between these members causing tubesheet clad separation and tube damage which has occurred in the industry. The auxiliary feedwater inlet sparger is provided with top discharge elbows to increase the drainage time for the sparger system in the event of auxiliary feedwater interruption.

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

Steam generator chemistry can be maintained through a combination of control of impurities delivered to the steam generator, monitoring and controlling the chemical environment within the steam generator, 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 can be maintained at a low level. A chemistry control program can be 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 can be utilized for the secondary systems. This method of secondary chemistry control helps to preclude 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 can 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 can be specified. Continuous sampling of and chemical addition to the steam generator monitors the effectiveness of feedtrain impurity controls and maintains a chemical environment conducive to low corrosion rates within the steam generator. Chemical sampling connections on the System 80 steam generator are located in both the recirculating downcomer water and the blowdown piping adjacent to the blowdown nozzle. These locations permit the separate evaluation of secondary fluid chemistry in the recirculated water and in the water within the region of the tube bundle containing the hottest tubes. Comparison of the

samples permits optimization of feedwater chemistry so that corrosive conditions can be avoided. Finally, steam generator blowdown, as described above, 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 can be minimized.

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. 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 System 80 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, continous 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. There is time available for operator action.

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. Equipment involved is accessible.

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. Procedures are consistent with normal DHR procedures.

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 C-E. 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 difficulty 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 C-E by NRC Division of Risk Analysis (DRA) attempted to demonstrate that the C-E 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.

In this report, the authors characterized their work as being a "back of the envelope" probabilistic risk assessment. At a March 16, 1982 meeting of the ACRS Subcommittee on the CESSAR application, one of the authors appeared and significantly qualified the conclusions of the draft PRA. The author noted that the PRA recommendations were "overstated" and recommended that the addition of PORV's should not be made a requirement for plants currently in the licensing process.

The NRC staff reconsideration of this PRA was prompted, in part, by C-E concerns presented in a letter dated March 4, 1982 from A. E. Scherer to D. G. Eisenhut. Since the author has revised the conclusions of the the DRA study, C-E comments on the draft PRA are presented again here only for completeness of the record.

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 treatment 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 the System 80 design.

   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, none of the current designs for System 80 plants have AFWS with 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 C-E plants and the specific AFWS designs.

b. Loss of Offsite Power

The results of this analysis indicate that the System 80 plants are acceptable as long as both motor driven AFW pumps can be powered by diesel generators. This is the case for all System 80 plants. Therefore the frequency of core melt resulting from loss of offsite power is 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 C-E designed HPSI systems. The C-E designs are simpler and are more reliable than those evaluated for Surry, Oconee, et al. The C-E 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 a C-E plant is much less than 10^{-4}. It seems inappropriate to draw conclusions on C-E designed systems from the results of analyses on non C-E 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 approximately 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 analyses.

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 inappropriate 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 is 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 to System 80 owners. Specifically:

   a. Inappropriate average core melt frequencies are used in 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, C-E has conducted a review of the System 80 design and has determined the following:

1. The System 80 NSSS will be coupled with highly reliable emergency feedwater systems, by addition of an interface requirement that the EFWS have an unavailability in the range of 10^{-4} to 10^{-5} per demand.

2. The System 80 NSSS is capable of achieving cold shutdown conditions using only safety grade systems, even without offsite power and with an added single failure.

3. The System 80 steam generator includes many features which will assure adequate tube integrity, minimizing concerns associated with operating reactors.

4. Even if all auxiliary feedwater supply were somehow lost, the secondary side of the steam generators could be depressurized to allow use of low head pumps which might be aligned to provide water to the steam generators from a number of sources.

5. Contrary to the probability analysis developed by DRA, installing PORV's will not result in a significant improvement in safety. The added costs are not justified.

Based upon the considerations listed above, C-E has concluded that the current System 80 design, strenghtened by addition of the interface requirement on reliability of the EFWS, provides adequate protection for the health and safety of the public. Continued licensing of the design is fully justified while responses are being prepared to meet the NRC request for additional information associated with rapid depressurization and decay heat removal capability.