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April 5, 1988

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555

Gentlemen:

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

Reference: Letter, Mr. M. O. Medford to NRC, dated April 3, 1988;
Subject: same as above.

Enclosure: Revised Justification for Continued Operation with
Combustion Engineering Steam Generator Tube Plugs,
dated April 5, 1988

The purpose of this letter is to transmit for your information a revision of the Justification for Continued Operation (JCO) which was initially submitted by our letter dated April 3, 1988 to make it consistent with the Nonconformance Report (NCR) No. 2-2367. As discussed in our conference call with Mr. Don Hickman during the morning of April 5, 1988, the safety evaluation accompanying the Nonconformance Report (NCR) No. 2-2367 has been incorporated in the JCO in its entirety. Also addressed in this letter is a discussion of the management philosophy regarding the continued operation of either Unit 2 or 3 in the event of a steam generator tube leak.

The Southern California Edison Company response to a tube leak in a San Onofre Unit 2 or 3 steam generator is designed to assure that leakage will not exceed the Technical Specification limit (720 gpd) while simultaneously ensuring that regular tube sheet inspection techniques can be relied upon to identify the leaking tube(s). It is the policy of the Edison Company, when a calculated leakage rate approaches 100 gpd, to evaluate the rate of increase of leakage, including the measurement uncertainty, and the specific history of the leak. The evaluation is then used to schedule a planned outage for the affected unit. The unit is then removed from service in accordance with this schedule. The progress of any steam generator tube leak is continually monitored and, should conditions change, the outage schedule is reevaluated and modified, as required.

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Mr. Hickman also requested during the aforementioned telephone call that SCE address plans with respect to future activities on Unit 2 relative to rolled C-E plugs and steam generator tube inspections which will be performed during the upcoming Unit 3 refueling outage. This information will be provided by separate correspondence within 30 days of the date of this letter.

If you have any questions or would like additional information, please do not hesitate to contact me.

Very truly yours,

M. O. Medford

Enclosure

cc: D. Hickman, NRR Project Manager, San Onofre Units 2 and 3
J. B. Martin, Regional Administrator, NRC Region V
F. R. Huey, NRC Senior Resident Inspector, San Onofre Units 1, 2 and 3

JUSTIFICATION FOR CONTINUED OPERATION WITH COMBUSTION ENGINEERING STEAM GENERATOR TUBE PLUGS

SAN ONOFRE NUCLEAR GENERATING STATION UNITS 2 AND 3

I. BACKGROUND

The purpose of this report is to provide the information necessary to allow continued plant operation with the existing Combustion Engineering (C-E) mechanical steam generator tube plugs.

The C-E mechanical tube plugs were installed in Unit 2 in January 1985 and in Unit 3 in February 1985. During the October 1985 Unit 3 Cycle 2 refueling outage, water was discovered behind several tube plugs. The apparent cause was improper rolling during installation, and the corrective action was to properly reroll each plug. No immediate action was required for Unit 2, and rerolling was planned for the next Cycle 3 refueling outage. During the March 1986 Unit 2 Cycle 3 refueling outage, several tube plugs were discovered to be missing from tubes in the steam generator E089 hot leg prior to the rerolling effort. All other plugs were verified to be in the proper location and each tube plug was properly rerolled. The missing tube plugs were replaced and properly installed. Appendix C of reference (a) documents the safety evaluation used to allow continued operation with the properly rerolled tube plugs. The conclusions stated that the missing plug would not represent a substantial safety hazard to the reactor coolant system.

On March 1, 1988, a primary to secondary leak developed on Unit 2, which ultimately resulted in a plant shutdown on March 17, 1988. The maximum calculated leak rate at shutdown was approximately 500 gallons per day (gpd) (technical specification limit is 720 gpd). Investigation of the primary to secondary leak identified a missing C-E mechanical plug in the cold leg of steam generator E088. This leaking tube was located in the active batwing wear region. Further inspections identified another missing plug in the cold leg of steam generator E089. This second tube did not have a through-wall defect.

An investigation was initiated to determine the cause and consequences of having C-E mechanical plugs dislodged from their plug location.

II. MARCH 1988 STEAM GENERATOR LEAK INSPECTION

A 600 pounds per square inch (psi) leak test was performed in steam generator E088 to locate the leakage source. Before the final pressure was attained, the source of the leakage was identified to be the cold leg side of tube number Row 43 Column 89. This tube is in the active batwing wear region and had previously been plugged with a C-E mechanical plug. This plug was missing and was not located in the cold leg channel head. This tube was replugged with a Westinghouse mechanical plug, and another 600 psi leak test was performed. During this leak test all tube plugs in both the hot and cold leg were verified to be in place. One other leak was identfied in a welded tube plug and this leaking weld plug was repaired.

Review of available steam generator E088 documentation from the Unit 2 Cycle 3 and Cycle 4 refueling outages identified that the plug had been forced out sometime during Cycle 3 operation. This condition was not discovered during the steam generator work conducted during the Unit 2 Cycle 4 refueling outage. Based on this, the available documentation for steam generator E089 was reviewed to ascertain if any additional plugs were missing. This review identified that only one additional tube plug was missing in the cold leg of steam generator E089.

A similar effort to that conducted in steam generator E088 was conducted in steam generator E089. All tube plugs were verified to be in place during a 600 psi leak test. One leaking C-E mechanical plug was identified in the hot leg. This plug was removed and replaced with a Westinghouse mechanical plug.

In summary, all plugs have been verified to be in place. In addition, those plugs in tubes with through-wall defects have been verified to be leak tight at 600 psi.

III. POSSIBLE FAILURE SCENARIO OF C-E STEAM GENERATOR TUBE PLUGS

To identify the possible failure scenario that is responsible for plug dislodgement, a root cause analysis was performed. After examining the operational history, qualification history, qualification test results, and mechanical design features of C-E mechanical plugs, many failure scenarios were hypothesized. However, only two of them cannot be refuted by a stress analysis that determines the internal tube pressure for various scenarios. These two scenarios are described as follows:

1. Before the re-roll of these two missing plugs during the Cycle 3 outage the tubes had been filled with water without any void. The water inside the tube was a result of a very small in-leakage over a long period of time. The re-roll completely sealed the plug and trapped the water. The plug was later ejected during subsequent plant heat-up with a high internal tube pressure caused by thermal expansion of trapped water.

2. The first roll of the two missing plugs was defective. During the operation of Cycle 2, the iron band on the plug was slightly corroded. The subsequent re-roll of these two plugs did not result in a complete "cold welding" condition as expected. In other words, the slightly corroded area in the iron band was not bonded by fusion of metal, but by contact stress. During subsequent Cycle 3 operation, the contact stress was relaxed and allowed water to leak into and fill up the tube. The plug was ejected after a plant trip

during which the secondary steam temperature increased by as much as 25 degrees F in 10 seconds. The internal pressure of the tube could be as high as 5,600 psi.

Both failure scenarios are possible and cannot be refuted by analysis and evidence. The main difference between these two scenarios is that the second scenario does not depend on the hypothesis that after the re-roll, these two tubes were filled with water. This solid water hypothesis is judged to be possible but not very likely, since after the primary side depressurization the water accumulated in these two tubes should be able to leak out. Any small void generated in these tubes would significantly reduce the subsequent pressurization.

By examining these two possible failure scenarios, we conclude that the probability of recurrence, i.e., tube plug dislodgement is very low. This conclusion is reached based on the fact that both scenarios require the coincidence of many factors. The first scenario requires (1) defective first roll, (2) complete seal by second roll with tubes filled with water, and (3) unbreached tubes. The second scenario requires (1) defective first roll, (2) defective second roll, and (3) unbreached tubes. Because all the re-rolled plugs (except for the two missing plugs) have gone through a heatup cycle and many plant trips during Cycle 3 without failure, it is reasonable to conclude that none of the plugs meets the requirements stated in the two possible failure scenarios.

The San Onofre experience with batwing wear (Reference [c]) and leaks caused by improper tube annealing confirms that leak-before-break is the only current tube failure scenario. Therefore, consequences of any potential future failures of the C-E plugs is bounded by the leak-before-break analysis.

IV. SAFETY EVALUATION

1. The probability of occurrence of an accident or malfunction of any equipment previously evaluated in the FSA will not be increased. The following analysis describes the basis for this answer.

With a possibility of multiple plugs dislodged from the steam generators, any change to the probability of following accidents and malfunctions of following equipment is considered:

(a) Flow blockage of a fuel channel

(b) Excessive fuel failure

(c) RCS pressure vessel cladding damage

(d) Core support structure damage

(e) Control rod/core upper guide structure damage

(f) Reactor coolant pump damage

(g) Damage to systems connected to RCS, including CVCS (chemical and volume control system), shut-down cooling system/safety injection system, and pressurizer spray valve.

Each of the items are discussed in detail below.

(a) Flow Blockage

The effect of partial blockage on DNBR has been included in the experimentally determined CE-1 CHF correlation. This conservatism is included in CE's design of reactor protection system. Since the partial blockage caused by multiple plugs is considered much less severe than that included in the CE-1 CHF correlation, the DNBR safety limit and DNBR related AOO's are not affected. Moreover, since this conservatism is always included in the SONGS Unit 2's reactor protection system, the probability of flow blockage is, in essence, assumed to be 1.0. As such, the probability of flow blockage will not be further increased with multiple loose plugs blocking the inlet of the flow channels.

It should be noted that CNB tests with a flow blockage at the inlet to several adjacent subchannels showed that the blockage did not affect (within the bounds of measurement uncertainty) the DNB performance of the fuel rods. The tests were conducted by CE with a 21 rod, one tube 5 x 5 test section at the Columbia University Heat Transfer Facility. The test section had a 48" heated length with a non-uniform radial power distribution and a uniform axial power distribution. The inlet to 11 of the 34 subchannels in the test section was completely blocked, reducing the inlet flow area by 36%. Two series of tests were run on the test section, one series with the inlet blockage in place and a second series with the blockage removed. Comparison of the measured Critical Heat Flux (CHF) to cause DNB for the blocked test series with those of the unblocked series showed that there was no effect on the bundle heat transfer capability due to the inlet blockage. Thus, because of the open lattice of rods, flow was able to redistribute quickly enough downstream of the blockage to preclude premature DNB.

(b) <u>Excessive Fuel Damage by Debris Generated by Loose Plugs</u>

An intact plug would not be able to reach the fuel since its dimensions will not allow it to pass through the most restrictive core internal orifice: i.e., the 0.422 inch diameter flow holes in the lower end fittings of the fuel assemblies. Fragmentation of the loose plugs will not occur based on the following evaluation:

The CE mechanical tube plug is fabricated from Inconel 690. At room temperature and at elevated temperature, Inconel 690 displays high yield and ultimate strengths along with good ductility. Inconel 690 has a high degree of metallurgical stability, forming no embrittling phases during long time exposures to elevated temperatures.

At the final phase of fabrication, the tube plug is annealed at approximately 1800 degrees Fahrenheit in order to obtain a soft iron band. This also puts the tube plug in the annealed condition without going below the minimum yield strength. The high temperature tensile properties for Inconel 690 for both cold-worked and hot-worked products indicate a ductile elongation of 45% at 600 degrees Fahrenheit.

A hardness traverse across a section of the wall of a fabricated tube plug at the major diameter shows a hardness of 96 Rb to 99 Rb (216 YHN to 234 YHN). The hardness is relatively consistent from the O.D. to the I.D. surfaces. The hardness of a tube plug that has been installed is slightly higher in the rolled zone due to the cold working of the rolling operation. The hardness in the rolled zone near the plug O.D. is 97 Rb to 100 Rb (222 YHN to 240 YHN), while the hardness near the I.D. surface is 31 Rc to 33 Rc (309 YHN to 326 YHN). It should be noted that this is just at the I.D. surface of the one-inch NCR rolled zone. The hardness at the tip of a tube plug is in the range of 91 Rb to 22 Rc (191 YHN to 247 YHN).

As a tube plug moves around in the primary coolant system, it may become dented or deformed from impact with primary component surfaces. This deformation should not significantly change the ductility of the tube plug. Extensive model testing of loose parts impinging on Inconel 600 Steam Generator tubes indicates that there was no significant loss of ductility as a result of surface damage inflicted by an impinging loose object. The percent of elongation changed from 35.8% for the unaffected surface to 34.7% for the peened surface. It is expected that Inconel 690 should behave in the same manner as Inconel 600 since they are similar materials.

Based on the ductile characteristics of the tube plug and on previous model testing of damage by loose parts impingement, it is concluded that a tube plug should not experience brittle fracture as a result of impact hardening associated with loose part impingement.

CE has had limited experience with loose plugs in Steam Generators. In all cases except Waterford 3 and San Onofre 2, this experience has been limited to the hot leg side where plugs have fallen out and been confined to the same channel head until their later discovery. Eleven loose plugs have been retrieved from hot leg plenums after some indefinite period of time being loose and subject to flow forces. In all cases, the plugs were found intact. There was no brittle fracture evident. Visual examinations made it apparent that the plugs had tumbled about in the turbulent flow. The plugs were scratched, dented and slightly out-of-round. On several plugs the thin iron band around the plug O.D. was no longer evident. In all likelihood, abrasive action removed the band and the remnants became an addition to the general inventory of corrosion deposits.

At Waterford 3, one loose plug was discovered on the top of the lower core support plate. This plug was originally installed in the cold leg plenum of #1 Steam Generator and probably came loose during one of the several plant depressurizations for maintenance conducted in the November to December, 1985 period. The loose plug was found in the region of highest coolant flow velocity (immediately below the lower end fitting of a fuel assembly) and highest neutron flux where an intact plug could be located. In addition, the plug had passed through one Reactor Coolant Pump. Upon visual examination, the plug showed evidence of denting and deformation, but it was entirely intact and showed no indication of breaking into small pieces.

Additional flow tests to simulate the behavior of loose plugs in a turbulent environment would not be expected to provide any additional or different data that has already been gathered by the twelve plugs which have been subjected to operational flow environments and then retrieved for examination.

All plugs which have been subjected to the flow tumbling actions of a loose part and then subsequently retrieved and examined showed no evidence of fragmentation.

Based on the above analyses, fuel failure due to loose Steam Generator plugs is not credible and, therefore, the probability of occurrence is not increased.

(c) RCS Pressure Vessel Cladding Damage

The internal surface of the reactor vessel lower head is clad with stainless steel and Inconel. The upper shell, intermediate shell, lower shell and lower portion of the bottom head were clad with the submerged arc welding process utilizing stainless steel material. A review of test data associated with single-pass process indicates that a nominal 0.1875 inch thick clad deposit with a hardness range of 82 Rb to 97 Rb (156 YHN to 222 YHN) is expected. The upper portion of the bottom head was clad with the submerged arc welding process utilizing an Inconel wire procedure. The Inconel clad has a nominal thickness of 0.2028 inches with a hardness range of 90 Rb to 95 Rb (185 YHN to 210 YHN).

The hardness of an as-built tube plug in the area of its major diameter is 96 Rb to 99 Rb (216 YHN to 234 YHN) and at the nose tip, 91 Rb to 22 Rc (191 YHN to 247 YHN). The lower hardness is near the O.D. surface and increases to the higher hardness as the I.D. surface is approached. These figures indicate that the major portion of a tube plug is slightly harder than either the Inconel or stainless steel clad.

The potential for wear could take the form of fretting wear, as when two contacting surfaces undergo very small oscillatory movements, or abrasive wear where material is removed from a surface by the cutting or gouging action of hard particles.

It is unlikely that a tube plug could wear a hole in the clad by fretting. The postulated worst case fretting wear situation would be one where the complete volume of a tube plug would wear away an equal volume of stainless steel clad in the smallest possible surface area. If one considers the cross-section of the major diameter of the tube plug positioned parallel to the cladding surface and projects in onto the clad surface as shown in Figure 1, this worst case scenario would be represented. Here, the cross-sectional area of the plug is 0.1219 square inches (0.1875 inch clad thickness times 0.650 inch plug diameter). This indicates that when the plug is completely worn away, there should still be 0.0364 square inches of clad thickness of 0.0560 inches. If the fretting wear were to occur against the Inconel clad surface, the amount of cladding remaining after the plug is completely gone should be 0.0713 inches. It should be noted that any amplitude of oscillation would cause this wear to be spread over an even larger clad area, thus reducing the depth of wear into the clad surface. In addition, since the plug was reduced in size due to equal loss of material, the forces normal to the wear plane should be reduced. This, in turn, should tend to further reduce the wear rate.

It is postulated that abrasive wear could occur if the plug were tumbling about in a random manner. In this case, the point of impact would be spread randomly over the entire reactor vessel lower head surface. No significant amount of abrasive wear should occur since successive impact points would not likely coincide.

It is concluded that fretting or abrasive wear through the reactor vessel cladding is not likely and, therefore, the probability of occurrence of this event will not be increased.

(d) Core Support Structure Damage

The components of the core plate and lower support assembly were reviewed and it was confirmed that all component thicknesses exceed the normal 0.1875 inch thickness assumed in Section c above for the reactor vessel cladding. The thinnest cross-sections are portions of the lower end fittings with a .213 inch minimum wall. However, this is a portion of only one of four support bosses on each lower end fitting. Based on the analysis done in Section c, approximately .132 inches would be worn away in the worst case by a loose plug, leaving .081 inches of metal in the local area. This wear in a highly localized area would not significantly compromise the strength of the lower end fitting since three additional support bosses exist and the lower end fitting insert pins would also support the fitting.

Based on the above review of the most limiting core internal component thickness and the analysis in Section c, it can be concluded that there will be no increase in the probability of occurrence of an accident or malfunction previously evaluated in the FSAR.

(e) Control Rod/Core Upper Guide Structure Damage

The analysis presented in Section b demonstrated that a loose steam generator tube plug is not likely to fragment and will remain intact. The most restrictive passage in the lower core internals is the 0.422 inch diameter flow holes in the lower end fittings of the fuel assemblies. The plugs have a major diameter of 0.668 inches, a nominal diameter of 0.650 inches, and a length of 5 inches. Even if the plug successfully negotiates the tortuous flow path from the reactor vessel inlet to the lower end fitting, it would be retained at this location and would not pass through the fuel to the upper internals and control rod area. Therefore, since the plug remains intact and is larger than the most restrictive orifice, it will not reach the upper internals nor the control rod area and the probability of occurrence of an accident or malfunction previously evaluated in the FSAR will not be increased.

(f) Reactor Coolant Pump Damage

The reactor coolant pumps are massive components with large flow passages. Due to the flow turbulence and large fluid forces in the pump, a loose plug would most probably pass through the pump impeller and casing area and not be retained in the pump. The plug dimensions are larger than the impeller to pump clearances precluding the plug from lodging in any pump interfaces. Additionally, the high fluid flow forces present in the pump impeller suction area will draw objects toward the center of the impeller where the flow velocity, thus the lifting force, is the highest. Therefore, the probability of occurrence of an accident or malfunction previously evaluated in the FSAR will not be increased.

(g) Damage to Systems Connected to the RCS

The following systems are connected to the RCS cold leg between the steam generator and the reactor vessel.

(1) Reactor Coolant Drain Lines

(2) RCS Letdown Line to CVCS

(3) Pressurizer Spray Valves

(4) Safety Injection/Shutdown Cooling Return

(5) CVCS Charging Line

In all of the above cases, the fluid flow forces in the RCS cold leg forcing a loose plug to the reactor vessel would dominate over forces tending to pull the plug into the above lines. Each case will be discussed separately.

(1) The reactor coolant drain lines are normally isolated from the RCS by two series of manual isolation valves and a pipe spool. Should the plug enter these lines, it would migrate to the first isolation valve and settle at this location. It would have no operational impact at this point since this line is not in service. Should it be necessary to drain the RCS through this line and the plug became lodged in the valve, this condition would be noted by the operator on manually closing the valve on startup. Such discovery would necessitate disassembly of the valve and plug removal. Should the plug migrate down this line, it would end up in the NSR reactor coolant drain tank and not affect any important to safety component.

(2) The major components of the CVCS letdown line inside containment include flow isolation valves 2TV0221 and 2HV9204, the regenerative heat exchanger and containment isolation valve 2TV9267. Should the plug enter this line, the first component it would encounter and most probably engage in, would be 2TV0221. This valve is one of two isolation valves which close on SIAS. The presence of the plug may prevent this valve from closing. 2HV9204 also receives a SIAS isolation signal and would close. As further backup for this valve containment isolation valve 2TV9267 may be closed manually (as well as outside containment isolation valve 2HV9205). Therefore, the multiple redundant components in this system would assure isolation of this line even should one valve be rendered inoperable by the loose plug. Should the plug pass through the letdown line and its major component valves and heat exchangers, it would be caught by the letdown filter before reaching the safety related charging pumps further down the line.

(3) There are two pressurizer spray valves in the RCS. Should a loose plug reach one of these valves and lodge in an open valve, it may preclude the valve from closing completely. This condition is compensated for by the backup heaters which would energize to add thermal input to the pressurizer. Should the heaters not be able to compensate for the flow through the open valve, RCS pressure would begin to drop ultimately reaching the RCS low pressure trip setpoint. This event is considered an anticipated operational transient within the plant design.

(4) The safety injection/shutdown cooling return line is not in service with the plant in operation or in startup. These lines have no flow through them to force a plug into them and they connect to the RCS at the top of the RCS cold leg thus precluding a plug from falling into the line during stagnant conditions in the RCS. Thus a loose plug will not enter the safety injection/shutdown cooling return line.

(5) The CVCS charging line is normally inservice providing flow to the RCS. Since flow is from this line to the RCS, a loose plug would not enter this line. Should the plug drift to this line with the charging line out of service, on initiation of flow the plug would be forced into the RCS.

Based on the review of the above components connected to the RCS, the probability of occurrence of an accident or malfunction previously evaluated in the FSAR would not be increased.

The above evaluation has shown that the probablility of malfunction of any equipment previously evaluated in the FSAR has not been increased as a result of these loose plugs. The probability of accidents in the FSAR has not been increased as a result of these plugs. The probability of occurance of specific accidents listed in section 2. of this evaluation will not be increased as a result of these loose plugs.

2. The consequences of an accident previously evaluated in the FSA will not be increased. The following accidents evaluated in the SONGS 2/3 FSA were reviewed to make this determination.

   (a) Single Reactor Coolant Pump Shaft Seizure - FSA Section 15.3.3.1

   (b) Control Element Assembly Misoperation - FSA 15.3.13

   (c) Chemical and Volume Control System Malfunction - FSA 15.5.1.1

   (d) Small Break LOCA

      1) Primary Sample or Instrument Line Break - FSA 156.3.1

      2) Steam Generator Tube Rupture - FSA 15.6.3.2

   (e) Loss of Coolant Accident - FSA 15.6.3.3

   Section 1 of this evaluation concluded that the probability of these accidents has not been increased as a result of lost steam generator tube plugs.

   (a) Single Reactor Coolant Pump Shaft Seizure

   The FSA Section 15.3.3.1 states that the results of a single reactor coolant pump shaft seizure are bounded by the single reactor coolant pump sheared shaft event.

   The sheared shaft events result in a reactor trip before flow reversal can occur. Therefore, the plug will not retrace its path to the cold leg side of the steam generator. Since the testing performed by CE demonstrated that under varying flow conditions, DNBR was not effected in a partially blocked assembly, the consequences of this event are not increased as a result of the missing steam generator tube plugs.

(b) Control Element Assembly Misoperation

This event assumes a stuck CEA as a result of mechanical jamming of the CEA fingers or jamming of the gripper.

This event is bounded by the CEA withdrawal analysis described in FSA Section 15.4.1. The consequence of this event can only be increased by delaying the reactor protection system response time to trip. The loose plug cannot physically affect any of the RCS parameters measured by the reactor protective system; therefore, the response time of the protection system cannot be increased by the plugs.

(c) Chemical and Volume Control System Malfunction

The consequences of this inadvertent boron dilution event can only be increased by delaying the response time of the reactor protection system. The presence of loose steam generator plugs cannot physically affect any of the parameters measured by the protection system to generate a reactor trip during this event.

(d) Small Break LOCA or Steam Generator Tube Ruptures

The consequence of these events cannot be increased by the presence of loose plugs since no flow reversal within the RCS occurs. The plug will remain within the bottom reactor vessel. The presence of a plug will not affect the reactor protection system or ECCS from initiating as a result of a small break LOCA.

(e) Large Break LOCA

During the blowdown phase of a large break LOCA, the loose plug would be ejected out the break as a result of the very large RCS flow rates. The consequence of the LOCA will not be increased as a result of this missile since the physical arrangement of the RCS piping prevents missiles generated by a large break LOCA in one loop from impacting the intact loops.

3. The possibility of an accident which is different than any already evaluated in the FSA will not be created.

No. The potential of creating new accidents that are not analyzed in the FSAR is analyzed below. This analysis concludes that creation of these events is not credible.

More specifically, it was hypothesized that the following new events be created:

(a) Small and large break LOCA in conjunction with multiple plug dislodgement from breached tubes.

(b) MSLB or MFLB event in conjunction with multiple plug dislodgement from breached tubes.

(c) SGTR event in conjunction with multiple plug dislodgement from breached tubes.

The maximum differential pressures that will be applied across the plugs during the LOCA, MSLB, and SGTR events are estimated to be as follows:

(a) <u>Large Break LOCA</u>: differential pressure = 900 psi with a maximum secondary steam pressure of 900 psig and a minimum primary pressure of 0 psig.

(b) <u>Small Break LOCA</u>: differential pressure = 1700 psi with a maximum primary pressure of 2500 psig and minimum secondary pressure of 800 psig.

(c) MSLB/MFLB: differential pressure = 2500 psi with a maximum primary pressure of 2500 psig and a minimum secondary pressure of 0 psig.

(d) SGTR: differential pressure = 1700 psi with a maximum primary pressure of 2500 psig and a minimum secondary pressure of 800 psig.

The maximum primary-secondary differential pressure for these events is 2500 psi and the maximum secondary-primary differential pressure is 900 psi. Since a high primary-to-secondary differential pressure that develops across a plug in a breached tube, will not dislodge a tube and the 900 psi secondary-primary differential pressure is within the range of CE's qualification test (Appendix B, Qualification Testing of Combustion Engineering Mechanical Tube Plug, CENC 1792), it is determined that these hypothesized new events are not credible.

It should be noted that during these four limiting events, it is possible that a few more plugs from unbreached tubes will be dislodged. It is hypothesized that there are a few leaking rolled plugs in the unbreached tubes that cannot be identified by the hydrostatic test performed recently. These leaking plugs with very small holes allow water to accumulate in the unbreached tube during normal operation. Once the limiting accident occurs, the secondary steam temperature may quickly rise by as much as 25°F. This temperature rise will result in thermal expansion of the accumulated water by as much as 0.7%. With this volume increase an internal pressure of approximately 5,600 psi will be developed resulting in dislodgement of a few more plugs out of the unbreached tubes. For breached tubes, leaking plugs with small holes will not result in a high internal pressure because the thermal expansion induced volume change will be relieved via the breached tube hole.

The dislodged plugs in the unbreached tubes during these limiting events, or other similar types of events, will not result in increase of the probability and consequences of previously analyzed FSAR events (see answers to questions A and B). As a result, dislodgement of a few plugs in the unbreached tubes does not have any safety significance.

4. The margin of safety as defined in the basis for any Technical Specification will not be reduced.

   The following Technical Specifications were reviewed to determine which sections might be affected by the presence of loose steam generator plugs within the RCS. This review selected the following sections as candidates for further evaluation.

Section:	2.1.1	Reactor Core
	3.1.1	Shutdown Margin
	3.4.1.3	Movable Control Assemblies
	3.4.2.4	DNBR Margin
	3.4.2.5	RCS Flow Rate

A review of these sections determined that ability to detect whether or not a LCO condition exists is not affected by the presence of loose plugs. The ability to respond to the requirements of any LCO in the Technical Specifications is also not affected by the presence of a loose plug.

V. CONCLUSION

Based on the results of a root cause analysis and the safety evaluation it is concluded that continued operation with C-E mechanical plugs will not represent a safety hazard. This conclusion is reached based on the following key considerations:

1. According to the two possible failure scenarios, the probability of future recurrence is low.

2. The dislodgement of a few mechanical plugs during limiting accidents is possible. However, these plugs are in the unbreached tubes. As such, they will not result in an increase of the probability and consequences of limiting accidents.

3. If a mechanical plug does dislodge itself and the tube is subsequently breached, the leak-before break scenario together with a Tech Spec limit on steam generator leak rate will ensure the plant operates within the bounds of FSA.

VI. REFERENCES

(a) Combustion Engineering Report, "Summary Safety Evaluation Report for Loose Steam Generator Mechanical Tube Plugs at San Onofre Nuclear Generating Station," dated March 1988

(b) Letter from S. E. Weismantel (C-E) to B. Katz (SCE) dated March 31, 1988; Subject: Evaluation of Unplugged Tube Defects

(c) Combustion Engineering Report, "Evaluation of Steam Generator and Diagonal Spacer Strip Interaction and Wear," dated March 1985