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October 20, 1982

Director, Office of Nuclear Reactor Regulation
Attention: D. M. Crutchfield, Chief
Operating Reactors Branch No. 5
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Gentlemen:

Subject: Docket No. 50-206
Status of Unresolved Safety Issues
San Onofre Nuclear Generating Station
Unit 1

Your July 6, 1982 letter requested information on the status of nineteen unresolved safety issues (USIs) at San Onofre Unit 1. Accordingly, pursuant to Section 50.54 (f) of 10 CFR 50, enclosed is our report, "Status of Category A Unresolved Safety Issues at San Onofre Nuclear Generating Station, Unit 1." This report provides a description of each USI, a discussion of the NRC staff's generic bases for continued operation that apply to San Onofre Unit 1, and site specific considerations that further justify continued operation. This report demonstrates that each USI is being adequately addressed at San Onofre Unit 1 and continued operation will pose no undue risk to the health and safety of the public.

If you have any questions or require further information, please contact me.

Subscribed on this 20th day of October, 1982.

Very truly yours,

M. O. Medford

M. O. Medford
Supervising Engineer
San Onofre Units 2 and 3 Licensing

Subscribed and sworn to before me
this 20th day of October, 1982.

Agnes Crabtree

Notary Public in and for the County
of Los Angeles, State of California

OFFICIAL SEAL
AGNES CRABTREE
NOTARY PUBLIC — CALIFORNIA
PRINCIPAL OFFICE IN
LOS ANGELES COUNTY
My Commission Expires Aug. 27, 1986

Enclosure

A001

STATUS OF CATEGORY A

UNRESOLVED SAFETY ISSUES

AT

SAN ONOFRE NUCLEAR GENERATING STATION

UNIT 1

Status of Category A

Unresolved Safety Issues at

San Onofre Nuclear Generating Station, Unit 1

PURPOSE

The purpose of this report is to identify the current status of those unresolved safety issues identified in the scope below. As requested by an NRC letter dated July 6, 1982, the following information is provided:

(1) Whether the issue has been resolved at San Onofre Unit No. 1;

(2) if so, how the issue has been resolved; and

(3) if full resolution has not occurred, the interim measures that have been taken to assure that continued operation would not pose an undue risk to the public.

SCOPE

The following Category A unresolved safety issues are addressed by this report:

A-1	Waterhammer
A-2	Asymmetric Blowdown Loads on the Reactor Coolant System
A-3, 4, 5	Pressurized Water Reactor Steam Generator Tube Integrity
A-9	Anticipated Transients Without Scram
A-11	Reactor Vessel Materials Toughness
A-12	Fracture Toughness of Steam Generator and Reactor Coolant Pump Supports
A-17	Systems Interaction in Nuclear Power Plants
A-24	Environmental Qualification of Class 1E Safety Related Electrical Equipment
A-26	Reactor Vessel Pressure Transient Protection

A-31 Residual Heat Removal Requirements

A-36 Control of Heavy Loads Near Spent Fuel

A-40 Seismic Design Criteria - Short Term Program

A-43 Containment Emergency Sump Reliability

A-44 Station Blackout

A-45 Shutdown Decay Heat Removal Requirements

A-46 Seismic Qualification of Equipment in Operating Plants

A-47 Safety Implications of Control Systems

A-48 Hydrogen Control Measures and Effects of Hydrogen Burns on Safety Equipment

A-49 Pressurized Thermal Shock

USI A-1

WATER HAMMER

Task A-1 addresses the issue of water hammer in nuclear power plants. A water hammer is an intense pressure pulse in a reactor fluid system that has the potential to damage fluid system components. The causes of water hammer have been attributed to rapid condensation of steam pockets, steam driven slugs of water, pump startup with partially empty lines, and rapid valve motions. Because of the continuing incidence of water hammer events and the safety significance of the systems involved, Task A-1 was initiated to develop systematic review procedures and acceptance criteria for licensing reviews and operating reactor reviews.

Revision 2 of Task Action Plan A-1 states the bases for the NRC staff's conclusion that operation of LWRs can proceed with reasonable assurance that the health and safety of the public is protected while this task is being conducted. These bases include the low probability of a failure due to water hammer, currently installed redundant engineered safety features that would adequately limit the consequences of a postulated water hammer induced accident, and site specific interim modifications required of operating licensees and license applicants. We also consider a failure at San Onofre Unit 1 due to water hammer to be a low probability event and acknowledge that San Onofre Unit 1 has redundant engineered safety features that would adequately limit the consequences of a postulated water hammer induced accident. In addition, our site specific interim modifications are discussed below.

Site specific interim modifications concerning water hammer at San Onofre Unit 1 were requested by the NRC staff's letters dated May 13, 1975, September 2, 1977, May 25, 1979, and September 12, 1979. Our responses to these requests are documented in submittals dated July 14, 1975, December 27, 1977, June 18, 1979, and February 14, 1980. These submittals document extensive evaluations of the history, potential, and consequences of water hammer at San Onofre Unit 1 and design changes and procedural modifications undertaken by us to prevent water hammer events. One key procedural modification submitted by our February 14, 1980 letter limits steam generator feedwater addition to below 150 gpm during situations particularly conducive to steam generator water hammer. A Westinghouse study determined a feedwater flow rate below 150 gpm effectively reduces the potential for steam generator water hammer. This study is documented in Westinghouse Technical Bulletin NSD-TB-75-7 dated June 10, 1975.

The NRC staff provided a Safety Evaluation Report on steam generator water hammer at San Onofre Unit 1 by letter dated April 22, 1980. The SER was based on a technical evaluation by the NRC's consultant, EG&G Idaho, Inc. The technical evaluation cited the feedwater piping geometry and administrative controls at San Onofre Unit 1 as effective means for reducing the probability of water hammer. The effective horizontal length of main feedwater piping

adjacent to any steam generator is less than 32 inches. This arrangement limits the length of pipe available for steam to enter during periods of feedring uncovery and proportionally limits the energy of potential water hammers due to steam-water slugging. The SER concluded that a steam generator water hammer is not likely to occur and that continued operation will pose no undue risk to the health and safety of the public. The NRC staff also requested additional information which we submitted by letter dated November 25, 1980.

Subsequent to the Safety Evaluation Report, we committed to install flow meters on AFWS and main feedwater lines. These flow meters will help operators prevent a water hammer event from being initiated by excessive feedwater flow to the steam generators during conditions particularly conducive to water hammer. The AFWS flow meters, described in our October 16, 1980 submittal, are presently installed. The flow meters for the main feedwater lines, described in our March 6, 1981 letter, are scheduled to be installed during the present maintenance outage.

As discussed above, the generic bases justifying continued operation apply to San Onofre Unit 1. We have also implemented interim modifications such as design and procedural changes which further reduce the probability of a water hammer event. Therefore, San Onofre Unit 1 can continue to operate with no undue risk to the health and safety of the public.

USI A-2

ASYMMETRIC BLOWDOWN LOADS ON PWR PRIMARY SYSTEMS

Task A-2 was initiated to investigate the generic implications of asymmetric blowdown loads on PWR primary coolant systems. Asymmetric blowdown loads could be caused by postulated double ended, or guillotine, pipe ruptures at specific locations of the primary coolant system. These loads were recently identified during the licensing process of a specific facility. The concern of this program is that since other facilities did not include these loads as design basis events, they are also vulnerable to the unacceptable consequences of a guillotine pipe rupture. The purpose of Task A-2 is to develop acceptance criteria to assure operating reactors do not pose an unacceptable risk to the public due to asymmetric blowdown loads.

Revision No. 2 of Task Action Plan A-2 states that there is reasonable assurance that continued operation, pending completion of this task, does not constitute an undue risk to the health and safety of the public. This conclusion is based on the low probability of an initiating event of sufficient magnitude to seriously challenge the structural adequacy of the vessel support members and on inservice inspections of welds. Based on the large size of piping in question and the high quality control standards for nuclear plant piping, the rupture probability of pipes is estimated to be between 10^{-4} to 10^{-5} per reactor year in Task Action Plan A-2. In addition, a pipe break will cause a high load only if the break occurs at specific locations of the reactor coolant system, which further decreases the probability of an asymmetric blowdown load. Finally, inservice inspections in accordance with ASME Code Section XI provide assurance that flaws in the primary system piping will be identified before significant leakage problems develop. The piping in question at San Onofre Unit 1 is large diameter piping and was subjected to high quality control standards before operation. We also perform inservice inspections of piping as required by ASME Code Section XI. Therefore, the generic bases for continued operation apply to San Onofre Unit 1.

We have also actively participated in the Westinghouse A-2 Owners Group, which provided an additional basis for justifying continued operation. Westinghouse conducted studies concerning pipe fracture mechanics (documented in WCAP 9558 Revision 2) and weld material toughness (documented in WCAP 9787). These reports concluded that even under the worst possible combination of loadings including a safe shutdown earthquake, a realistically postulated flaw will not propagate around the circumference of the pipe and cause a guillotine break. Rather than propagate, a crack will stabilize, which will allow the leak detection system to identify the leak and alert the operator. Based on these two studies, the Westinghouse A-2 Owners Group proposed the "leak before break" resolution to this issue. A NRC Draft Topical Report Evaluation issued in March, 1981 indicated that leak before break is an acceptable resolution to this issue.

Revision 2 of Task Action Plan A-2 justifies continued operation based on the low probability of a structural support failure due to an asymmetrical blowdown load and on inservice inspections. We stated in the above discussion that these generic bases for continued operation apply to San Onofre Unit 1. The leak before break resolution to this issue is also discussed above. This resolution is based on studies that demonstrate a guillotine pipe rupture will not occur in a Westinghouse NSSS. Therefore, San Onofre Unit 1 can continue to operate with no undue risk to the health and safety of the public.

USIs A-3, 4, and 5

STEAM GENERATOR TUBE INTEGRITY

Tasks A-3, 4 and 5 address the degradation of steam generator tubes in PWRs. Tube degradation can be caused by corrosion-induced wastage, cracking, reduction in tube diameter (denting) and vibration induced fatigue cracks. The primary concern of these tasks is to assure degraded tubes can maintain adequate safety margins during normal operation and under accident conditions. Establishment of inspection and plugging criteria, administrative controls, and operating procedures are objectives of these tasks to assure adequate safety margins are maintained.

NUREG-0886, "Steam Generator Tube Experience," discusses the technical resolution of Unresolved Safety Issues A-3, A-4 and A-5. This discussion states that the NRC staff has concluded that continued operation and licensing do not constitute an undue risk to the health and safety of the public. This conclusion is based on the NRC's case-by-case evaluations of plants with adverse steam generator tube experience. These evaluations have resulted in plant specific corrective measures at many nuclear power plants, including San Onofre Unit 1. The following discussion identifies these corrective measures at San Onofre Unit 1 and demonstrates that continued operation poses no undue risk to the health and safety of the public.

Steam generator tube inservice inspections performed since 1972 at San Onofre Unit 1 have indicated that tube degradation has been caused by anti-vibration bar wear, denting (including lower tube support plate cracking and deformation), general phosphate wastage and IGA. Actions taken to monitor, control or eliminate these degradation mechanisms are discussed below.

1) During 1976-1977, new anti-vibration bars were installed to prevent further degradation at anti-vibration bar/steam generator tube interfaces. Our February 1, 1977 letter provided the details of this modification.

2) In 1978, Technical Specifications were implemented requiring routine inspections to monitor tube degradation induced by anti-vibration bar wear, denting and general phosphate wastage. The Technical Specifications delineate the criteria for selecting tubes to be inspected, the inspection frequency and the acceptance criteria for continued operation, including tube plugging limits. Since 1976, these inspections indicate that (1) significant or excessive degradation of tube integrity due to wastage type thinning imperfections is not occurring, (2) significant or excessive overall growth of tube wear at anti-vibration bar interfaces is not occurring and (3) significant or excessive progression in the magnitude of steam generator tube support plate inplane expansion or hourglassing and tube denting are not occurring. Our letter dated September 21, 1982 submitted the results of our most recent inspection.

3. Plugging limits and maximum primary-to-secondary leakage limits contained in the Technical Specifications ensure that (1) degraded tubing will retain adequate structural margins over the full range of normal operating, transient and postulated accident conditions and (2) cracks having a primary-to-secondary leakage less than the maximum limit during operation will have an adequate margin of safety to withstand the loads imposed by normal operation and postulated accidents. The plugging limits have been established consistent with the guidelines in Regulatory Guide 1.121.

4. During 1980-1981, sleeves were installed to span the region of the tubes affected by IGA. The details of the sleeving project were approved by the NRC staff as documented in their Safety Evaluation Report transmitted by letter dated June 8, 1981. As required by a license condition, the sleeved tubes and non-sleeved tubes in regions suspected to be conducive to IGA were inspected in 1982. The results of this inspection, submitted by our letter dated September 21, 1982, identified no degradation of sleeved tubes and only a few non-sleeved tubes with IGA indications. Future inspections will continue to monitor the status of IGA as discussed in our September 21, 1982 letter.

5. In 1981, the maximum primary-to-secondary leakage limit was reduced to provide increased margins against a potential gross tube failure caused by IGA. The details of this limit reduction were provided by our April 10, 1982 letter and incorporated into the Technical Specifications by NRC letter dated June 8, 1982.

6. In 1981 and 1982, primary and secondary hydrostatic testing of tube bundles were performed. The results of these inspections were provided in our letters dated April 10, 1981 and September 21, 1982. This testing will be continued during subsequent outages when tube inspections are scheduled.

7. In 1981, improved controls of secondary water chemistry, hot and cold steam generator soaks and reduced temperature operation were implemented to reduce the rate of further degradation. The details of these measures were provided by our April 10, 1981 and September 21, 1982 letters.

8. During 1980-1981, operational ALARA and audit programs were implemented to assure minimal exposures to plant personnel during steam generator and general work activities. These programs are administratively controlled by a dedicated plant organization.

9. In 1982, a foreign materials and loose parts inspection of the steam generators was performed. Foreign materials and loose parts identified though this inspection were retrieved and evaluated with

the results submitted to the NRC by our September 21, 1982 letter. Based on this inspection, a formal foreign material exclusion and inventory control program will be implemented as stated in our September 21, 1982 letter.

The measures discussed above are consistent with the measures outlined in NUREG-0886 which formed the bases for the NRC staff's determination that continued operation is justified. These measures assure that steam generator tube integrity is being adequately addressed at San Onofre Unit 1 and continued operation will pose no undue risk to the health and safety of the public.

USI A-9

Anticipated Transients Without Scram

Unresolved Safety Issue A-9 concerns anticipated transients without scram (ATWS) in nuclear power plants. An anticipated transient without scram can be described as a failure of the scram system to shut down the reactor following a disruption of normal heat dissipation paths. These disruptions are expected to occur during the life of the plant so the plant is accordingly designed to withstand their effects. In the event a required scram does not occur, the consequences of some postulated ATWS events are unacceptable. Task A-9 was initiated to evaluate ATWS events and recommend necessary changes to reduce the probability and the consequences of ATWS.

The Commission is currently reviewing three alternative rules which have been proposed for incorporation into Title 10 of the Code of Federal Regulations as Section 50.60. These proposed rules set standards for the reduction of risk from anticipated transients without scram for light water cooled power reactors. Until a final rule is approved, the Commission has concluded that interim operation is acceptable. This conclusion and the bases for this conclusion were stated in the November 24, 1981 issue of the Federal Register, 46-FR-57522:

> "The Commission believes that the likelihood of severe consequences arising from an ATWS event during the two to four year period required to implement a rule is acceptably small. This judgment is based on (a) the favorable experience with the operating reactors, (b) the limited number of operating nuclear power reactors, (c) the inherent capability of some of the operating PWRs to partially or fully mitigate the consequences of ATWS events, (d) the partial capability of the recirculation pump trip feature to mitigate ATWS events that has been implemented on all BWRs of high power level, and (e) the interim steps taken to develop procedures and train operators to further reduce the risk from some ATWS events. On the basis of these considerations, the Commission believes that there is reasonable assurance of safety for continued operation until implementation of a rule is complete."

We have reviewed the above bases for continued operation and determined all but Item (d) apply to San Onofre Unit 1. Item (d) does not apply since San Onofre Unit 1 is a PWR and Item (d) applies to BWRs only. Item (a) applies because San Onofre Unit 1 has never experienced an ATWS event and Item (b) applies because San Onofre Unit 1 is one of the limited number of operating nuclear power reactors. The November 24, 1981 issue of the Federal Register, 46-FR-57524, states that Westinghouse designed plants have a capability to mitigate nearly all ATWS events and already have a high level of safety. Since San Onofre Unit 1 is a Westinghouse designed plant, Item (c) applies. Item (e) applies because we have implemented emergency operator procedures addressing ATWS events. These procedures are based on the Emergency Response Guidelines (ERG) developed by Westinghouse.

The above discussion demonstrates that the NRC staff's bases that apply to Westinghouse PWRs and justify continued operation until resolution of this issue apply to San Onofre Unit 1. Therefore, San Onofre Unit 1 can continue to operate with no undue risk to the health and safety of the public.

USI A-11

REACTOR VESSEL MATERIALS TOUGHNESS

Unresolved Safety Issue A-11 concerns the reduction of reactor vessel materials toughness as a result of neutron irradiation. Nuclear reactor vessels designed to the most recent acceptance criteria as defined in ASME Code Section III are considered to have adequate safety margins against radiation induced brittle failure. However, plants designed prior to ASME Code Section III may have the potential for marginal fracture toughness within their design lives. The purpose of Task A-11 is to develop acceptance criteria for the reevaluation of reactor vessels fabricated before ASME Code Section III to determine if acceptable safety margins against failure will exist during the lifetimes of these plants.

Revision 2 of Task Action Plan A-11 states that continued operation until resolution of this issue will not adversely affect the health and safety of the public. We have reviewed the bases in the Task Action Plan for this conclusion and determined the bases discussed below apply to San Onofre Unit 1.

Pressure-temperature operating limits are imposed on licensees to assure reactor vessels are not exposed to high pressure in the material transition temperature regime. This precaution prevents vessel failure due to brittle fracture. However, neutron irradiation causes an upward shift in the material transition temperature. To assure pressure-temperature limits are based on current material properties, Regulatory Guide 1.99 requires a material surveillance program of the reactor vessel. The results of this program are then used to revise the pressure-temperature operating limits. This program assures the pressure vessel of San Onofre Unit 1 will not experience a failure due to brittle fracture. Revision 2 of Task Action Plan A-11 also cites studies of the toughness margins in operating reactors. These studies indicate adequate toughness margins can be maintained in operating plants under normal and accident conditions in the transition temperature region significantly beyond the scheduled completion of this task.

In addition to the generic arguments discussed above supporting safe interim operation of San Onofre Unit 1 and other nuclear plants, there are site specific assurances that San Onofre Unit 1 can continue to operate with no undue risk to the health and safety of the public. Two NRC evaluations of the reactor vessel at San Onofre Unit 1 have been conducted. The first evaluation was conducted in 1975-1976 and is documented in NUREG-0081, "Evaluation of the Integrity of Reactor Vessels Designed to ASME Code, Sections 1 and/or VIII." The second evaluation was conducted in 1978-1979 and is documented in NUREG-0569, "Evaluation of the Integrity of SEP Reactor Vessels." The scope of these evaluations included: design, materials, material surveillance programs, pressure-temperature operating limits, and inservice inspection programs. Generic safety items were also reviewed in the evaluation documented by NUREG-0569. In these reports the NRC staff concluded that the

design conservatisms and vessel fabrication materials of San Onofre Unit 1 are essentially in accordance with ASME Code Section III, which defines acceptance criteria for reactor vessels presently undergoing the licensing process. The reports also conclude that the combination of inservice inspections, conservative pressure-temperature operating limits, low vessel stresses, and the use of materials having acceptable fracture toughness properties provide that the vessel integrity will be maintained at acceptable levels throughout service life.

The generic bases supporting continued operation that apply to San Onofre Unit 1 are discussed above. These applicable generic bases include pressure-temperature operating limit revisions required by Regulatory Guide 1.99 and studies that demonstrate the existence of acceptable safety margins well beyond the expected completion of this task. The above discussion also cites two NRC studies of the reactor vessel at San Onofre Unit 1. These studies concluded that vessel integrity will be maintained at acceptable levels throughout service life. Therefore, San Onofre Unit 1 can continue to operate with no undue risk to the health and safety of the public pending resolution of this issue.

USI A-12

Fracture Toughness of Steam Generator and Reactor Coolant Pump Supports

During the licensing proceedings for North Anna Units 1 and 2, supplemental tests revealed that some steam generator and reactor coolant pump supports were fabricated from steel having low fracture toughness. Task A-12 was subsequently initiated to determine the generic extent of low fracture toughness steel use in PWR steam generator and reactor coolant pump supports. The issue of resistance to lamellar tearing was initially included in Task A-12 but later removed to allow further study before being addressed by the industry.

Task Action Plan A-12 states that, until resolution of this issue, continued operation of PWRs is justified based on the extreme conditions required for a support failure. A support failure is not expected to occur except under the unlikely combination of (1) an initiating event determined to be of very low probability (normal operating stresses are very low), (2) nonredundant and critical support member(s) of low fracture toughness (many supports contain redundant members), (3) member operating temperatures low enough that upper shelf energy absorption (where fracture toughness properties are best) is not reached, and (4) a flaw of such large size that the stresses imparted during (1) above would be of such intensity that crack arrest would not occur and the member(s) would fail in a brittle manner. The combination of factors necessary for a support failure discussed above are not expected to occur at San Onofre Unit 1 due to design conservatisms and operating procedures. Therefore, the NRC staff's conclusion that continued operation is justified based on the extreme conditions required for a support failure applies to San Onofre Unit 1.

We have also actively worked with the NRC staff to resolve this issue at San Onofre Unit 1. By letter dated September 15, 1978 we provided information regarding steam generator and reactor coolant pump support materials as requested by the NRC staff's letter dated October 4, 1977. The NRC subsequently issued NUREG-0577 for comment which provided their evaluation of the information submitted by the industry on steam generators and reactor coolant pump support materials. The industry submittals being evaluated by Franklin Research Center, which included San Onofre Unit 1, were not included in the NUREG-0577 evaluation. By letter dated July 23, 1980 the NRC staff requested additional information on San Onofre Unit 1 support material which we provided by letter dated September 23, 1980.

The NRC staff has subsequently conducted two meetings with the industry to discuss resolution of Task A-12. In the first meeting, held on August 27, 1980, the NRC staff informed the industry that resolution of the lamellar tearing issue was being deferred while the staff conducted additional research. The NRC staff confirmed the removal of lamellar tearing from Task

A-12 by letter dated October 6, 1980. In the second meeting, held December 17, 1980, the NRC staff indicated that their resolution of this issue would be provided in the final issue of NUREG-0577 to be published in May, 1981 and that industry wide resolution was expected as early as December 31, 1982. In addition, it was indicated that the SER for San Onofre Unit 1 would be available in early 1981. To date, the final issue of NUREG-0577 has not been published and the SER for San Onofre Unit 1 has not been received. Therefore, resolution of this issue is pending NRC action.

The NRC staff has justified continued operation until completion of this issue based on the extreme conditions required for a support failure. These same extreme conditions are required for a support failure at San Onofre Unit 1. Therefore, the generic basis for continued operation applies to San Onofre Unit 1. The above discussion also demonstrates our active participation with NRC staff efforts to resolve this issue. Based on the preceeding discussion, San Onofre Unit 1 can continue to operate with no undue risk to the health and safety of the public.

USI A-17

SYSTEMS INTERACTION IN NUCLEAR POWER PLANTS

Task A-17 involves the development of a systematic process to review plant systems to determine their impact on other plant systems resulting from normal operation and design basis accidents. The purpose of this task is to identify where the present design, analysis and review procedures may not acceptably account for potentially adverse systems interaction and to recommend the regulatory action that should be taken to rectify deficiencies in the procedures. The NRC anticipates that this task will confirm that current licensing requirements and procedures acceptably control the potential for adverse systems interactions, even though some modifications in the review procedures and licensing requirements may be made. This task is no longer being pursued as an unresolved safety issue but has been included in the TMI Action Plan as Item II.C.3, "Systems Interaction."

Revision No. 2 of Task Action Plan A-17 states that while this task is being performed, continued operation and plant licensing can proceed with reasonable assurance of protection to the health and safety of the public. We have reviewed the bases in the task action plan for continued plant operation and have concluded the following bases apply to San Onofre Unit 1: (1) The licensing design requirements originally followed for San Onofre Unit 1 provide reasonable assurance against potentially adverse systems interactions, (2) operating experience to date has demonstrated that San Onofre Unit 1 has been designed to provide reasonable assurance that adverse systems interactions will not occur, and (3) site specific corrective measures have been taken at San Onofre Unit 1 to further reduce the potential for adverse systems interactions. These site specific measures are discussed below.

Task Action Plan A-17 has identified the following issues as areas of concern: high energy line breaks, missiles, high winds, flooding, seismic events, fires, operator errors, and sabotage. Each of these issues is being addressed through various programs at San Onofre Unit 1 as discussed below.

San Onofre Unit 1 is one of the 11 operating plants being reviewed as part of the Systematic Evaluation Program (SEP). The following SEP topics address the specific concerns of Unresolved Safety Issue A-17: high energy line breaks (Topics III-5.A and III-5.B), missiles (Topics III-4.A, III-4.B, III-4.C, and III-4.D), high winds and tornados (Topic III-2), flooding (Topics II-3.A, II-3.B, II-3.B.1, and III-3.A) and seismic events (Topic III-6). These issues will be resolved in the integrated assessment of SEP. Resolution of SEP will assure that San Onofre Unit 1 maintains safety margins comparable to those required by current licensing criteria.

Fire protection requirements are specified in 10 CFR 50.48. Implementation of this rule has required extensive modifications to San Onofre Unit 1. The NRC staff's Safety Evaluation Report dated February 4, 1981 documents our approach

to implementing these modifications and identifies closed and open items. Closed items consist of modifications required to comply with 10 CFR 50.48(d). Most of these modifications have been completed while those modifications not presently installed are scheduled to be completed before the end of the present maintenance outage. Open items concern modifications and additions necessary to comply with 10 CFR 50.48(c) as specified in Item III.G of Appendix R to 10 CFR 50. Plans and schedules for compliance have been developed and preliminary engineering is underway; however, an exemption has been requested deferring the implementation schedule to be concurrent with SEP. This exemption request is under NRC review.

The final two issues, operator errors and sabotage, are also being addressed at San Onofre Unit 1. Accident evaluations are based on the single failure criterion. These evaluations review operator actions as well as physical systems to assure that a single failure does not initiate or aggravate an accident. As specified in Appendix C of 10 CFR 73, San Onofre has a security plan to prevent sabotage and other threats to the security of the plant. We provided the NRC staff with the latest revision to our security plan, "Physical Security Plan, San Onofre Units 1, 2 and 3," by our letter dated August 18, 1982.

We have reviewed the NRC staff's bases for continued operation pending completion of Task Action Plan A-17, now reorganized under TMI Action Plan Item II.C.3, and discuss above the bases that apply to San Onofre Unit 1. The above discussion also describes site specific activities concerning high energy line breaks, missiles, high winds, flooding, seismic events, fires, operator errors, and sabotage. These site specific activities add further assurance that adverse systems interactions will not occur at San Onofre Unit 1. Therefore, this issue is being adequately addressed and San Onofre Unit 1 can continue to operate with no undue risk to the health and safety of the public.

USI A-24

ENVIRONMENTAL QUALIFICATION OF CLASS 1E

SAFETY RELATED ELECTRICAL EQUIPMENT

Task A-24 was initiated to develop standardized practices for environmental qualification of safety related electrical equipment. Environmental qualification ensures that safety related electrical equipment can perform required safety functions in the event of an accident even though the equipment is exposed to a harsh environment. Prior to these standardization efforts, qualification was done on a case-by-case basis which resulted in a variety of qualification methods and documentation levels. In addition, qualification acceptance criteria have been periodically revised so that many plants have equipment qualified to different standards.

The NRC's letter to SEP plants dated February 15, 1980 and IE Bulletin 79-01B directed operating plants to address the environmental qualification of safety related electrical equipment. With the February 15, 1980 letter the NRC transmitted the staff's DOR guidelines on equipment qualification. In addition NUREG-0588, which provided the staff's position on IEEE 323-1974 and 1971, was issued. The NRC Commission then issued their Memorandum and Order dated May 23, 1980 which required that all safety-related electrical equipment be qualified to the DOR Guidelines or NUREG-0588 by June 30, 1982. This deadline has since been rescinded and left open by the Commission until approval of a final environmental qualification regulation.

By letters dated June 18, 1980 and October 31, 1980 we provided the NRC with a list of equipment that required environmental qualification, the status of each equipment's qualification and justification for continued operation. The NRC staff transmitted their Safety Evaluation Report which included Franklin's Technical Evaluation Report by letter dated June 2, 1981. We responded to this SER on November 4, 1981. Our response provided an update of the environmental qualification program and indicated that some equipment had been replaced or refurbished to ensure operation following an accident. In a meeting held with the NRC on February 17, 1982 we were requested to provide information justifying San Onofre Unit 1's continued operation. That justification was transmitted to the NRC by letter dated February 24, 1982. The NRC concluded that continued operation of San Onofre Unit 1 was justified in their March 8, 1982 letter.

As discussed above, the NRC staff provided specific guidelines for environmental qualification of safety related electrical equipment. Based on these guidelines we initiated a review of the electrical equipment at San Onofre Unit 1. Component by component justifications for continued operation were submitted to the NRC. The NRC's March 8, 1982 letter indicated that they agreed with the bases and conclusions of our submittals. Therefore, continued operation of San Onofre Unit 1 will pose no undue risk to the health and safety of the public.

USI A-26

REACTOR VESSEL PRESSURE TRANSIENT PROTECTION

Task A-26 concerns overpressurization transients in nuclear power plants that could violate technical specifications based on 10 CFR 50 Appendix G regulations. The majority of these overpressurization events have occurred while the reactor is water solid and at relatively low temperature. The lower material toughness at these temperatures decreases the margin of safety to brittle fracture. Increased neutron irradiation also decreases safety margins by reducing material toughness. The immediate concern, therefore, is older operating facilities. The purpose of Task A-26 is to identify necessary actions that will assure adequate overpressurization protection is provided at nuclear power plants.

Revision 1 of Task Action Plan A-26 states that continued operation of nuclear power plants will not present an undue risk to the health and safety of the public pending resolution of this issue. This decision is based on short term and long term measures that have been required of operating reactor licensees. Short term required measures include procedural changes, administrative controls, and installation of pressure alarms in the control room. The long term required measure is installation of an Overpressurization Mitigating System. Our responses to these requirements as they relate to San Onofre Unit 1 are discussed below.

The specific short term measures that have been required of licensees are: (1) upgrading of operating procedures to alert plant operators to the potential for pressure transients, (2) minimization of the time during which the plant is in a water solid condition, (3) deenergization of high head pumps not required during cold shutdown, and (4) installation of an alarm that will alert the operator if the system pressure approaches the limits of Appendix G to 10 CFR 50. Our responses to these requirements were transmitted by our letters dated April 22, 1977, May 2, 1977, October 12, 1977 and February 23, 1982.

The Overpressurization Mitigating System (OMS), described in our letter dated October 12, 1977, was completed during the Fall, 1978 refueling outage. The OMS utilizes the two independent pressurizer power operated relief valves (PORVs). The PORVs have been equipped with low pressure activation logic in addition to the already existing high pressure activation logic. If the low pressure setpoint is exceeded while the OMS is enabled, the PORVs automatically open and an annunciator warns the operator that an overpressurization transient is in progress. The relief capacity of each PORV is sufficient to mitigate any credible overpressurization transient.

There are also administrative controls and design features in addition to the short term measures and the OMS described above which further assure the prevention and mitigation of overpressurization events at San Onofre Unit 1.

Administrative controls govern plant cooldown and heatup, pump starts and stops, and system testing. Design features such as pressure relief paths through normal letdown lines and residual heat removal inlets provide a mitigation capability in addition to the OMS. These administrative controls and design features are described in our previously identified letters dated April 22, 1977, May 2, 1977, October 12, 1977 and February 23, 1982.

We have reviewed the NRC staff's bases for continued operation pending resolution of this issue as stated in Task Action Plan A-26, Revision 1. These bases consist of implementation of short term measures and installation of an Overpressurization Mitigating System. We identified documentation in the above discussion that demonstrates our compliance with the short term measures and installation of the OMS. The same documentation also identifies additional administrative controls and design features which further assure prevention and mitigation of overpressurization events. Therefore, San Onofre Unit 1 can continue to operate with no undue risk to the health and safety of the public.

USI A-31

RHR SHUTDOWN REQUIREMENTS

Task A-31 was initiated to propose acceptance criteria for RHR systems to assure a reactor can achieve and maintain cold shutdown from a hot standby condition. The resulting acceptance criteria were incorporated into Section 5.4.7 of the Standard Review Plan and Branch Technical Position RSB 5-1. Staff positions on design requirements for residual heat removal systems were also incorporated into Regulatory Guide 1.139, "Guidance for Residual Heat Removal."

To assure older operating plants, including San Onofre Unit 1, adequately address the concerns of Task A-31, the Systematic Evaluation Program (SEP) includes a review of the RHR system under SEP Topics V-10.B and V-11.A. This review will compare the existing RHR system with acceptance criteria stipulated in Section 5.4.7 of the Standard Review Plan, Branch Technical Position RSB 5-1, and Regulatory Guide 1.139 and determine any necessary modifications. The current status of this review is documented in the Safety Evaluation Reports discussed below.

The NRC staff's letter dated August 3, 1981 transmitted the Safety Evaluation Report for SEP Topic V-11.A, "Electrical, Instrumentation, and Control Features for Isolation of High and Low Pressure Systems." The primary concern of this topic is to assure that the low pressure systems, which includes RHR, are not subjected to coolant pressures that exceed design limits. Based on the review of Topic V-11.A, the NRC staff does not believe that the RHR system should be modified (if at all) until the low temperature overpressurization protection review (USI A-26) is completed. A detailed justification for continued operation until USI A-26 is resolved is presented in this report in the section: "USI A-26 - Reactor Vessel Pressure Transient Protection."

The Safety Evaluation Report of safe shutdown systems for San Onofre Unit 1 was transmitted to us by letter dated June 20, 1981. This SER included the review of SEP Topic V.10-B, "RHR Reliability." The NRC staff's June 20, 1981 letter requested that we review the SER and inform them if the as-built facility differs from the licensing basis assumed in the assessment. We provided this information and additional comments in our letter dated April 26, 1982. Completion of the integrated assessment phase of the SEP review will assure resolution of Task A-31 concerns.

Task A-31 efforts resulted in revisions to RHR system acceptance criteria. SEP Topics V-10.B and V-11.A were developed to reevaluate RHR systems of older operating plants based on the new criteria and to recommend necessary changes. This review has not been finished for San Onofre Unit 1. However, based on our efforts to resolve NRC staff questions concerning the SEP reviews and the NRC staff's interim positions contained in the Safety Evaluation Reports discussed above, the concerns of Task Action Plan A-31 are being adequately addressed at San Onofre Unit 1. Therefore, continued operation will pose no undue risk to the health and safety of the public.

USI A-36

CONTROL OF HEAVY LOADS

Task A-36 was established to review the adequacy of current licensing criteria and safety measures associated with heavy load handling at nuclear power plants. The concerns of this issue are the unacceptable consequences of a potential heavy load drop at specific plant locations. A drop in the spent fuel pit or reactor vessel could result in a radioactive release to the environment while a drop in other locations could damage safety related equipment. Resolution of this issue will establish minimum criteria for operating plants and reduce the potential and consequences of postulated heavy load drop accidents to an acceptable level.

Revision No. 2 of Task Action Plan A-36 states that the likelihood of a heavy load handling accident which damages enough fuel to result in unacceptable consequences will be small while the staff's generic evaluation proceeds. For plants licensed before current acceptance criteria, this judgment is based on design conservatisms, administrative controls, and specific interim measures. As discussed below, these bases apply to San Onofre Unit 1.

San Onofre Unit 1 was issued a provisional operating license in January of 1968. To date there has never been a heavy load drop at San Onofre Unit 1. This history of safe load handling verifies the adequacy of design conservatisms and administrative controls at San Onofre Unit 1.

The NRC's letter dated December 22, 1980 requested us to implement specific interim measures based on NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants." Our letters dated February 5, 1982, February 22, 1982, and April 9, 1982 provided our responses to these interim measures. These responses included procedural changes, administrative controls, personnel training programs and inspection programs.

The NRC's December 22, 1980 letter also requested that we conduct a review of San Onofre Unit 1 and implement necessary modifications to comply with criteria stipulated in NUREG-0612. The review was requested to be conducted in a six month phase and a nine month phase. Our six month report was submitted by letter dated April 1, 1982 and our nine month report was submitted by letter dated July 6, 1982. The NRC staff issued a Draft Technical Evaluation Report (TER) by letter dated August 3, 1982 which provided a preliminary evaluation of our interim measures and six month report. Information required to resolve the open items contained in the Draft TER will be provided in October, 1982.

With respect to heavy load handling, the NRC staff has justified continued operation of plants licensed before current acceptance criteria based on design conservatisms, administrative controls, and interim requirements. In

the above discussion we cite evidence of the adequacy of design conservatisms and administrative controls at San Onofre Unit 1. We also cite documentation that demonstrates we have implemented the interim requirements stipulated in the NRC's December 22, 1980 letter. Also in response to this letter, we are conducting a long term review of heavy load handling at San Onofre Unit 1 as documented in our six month and nine month reports. This discussion demonstrates that the generic bases for continued operation apply to San Onofre Unit 1 and that we are adequately addressing the long term resolution of this issue. Therefore, continued operation will pose no undue risk to the health and safety of the public.

USI A-40

SEISMIC DESIGN CRITERIA - SHORT TERM PROGRAM

The current seismic design criteria stipulated in NRC regulations and Regulatory Guides assure that safety related systems can withstand the effects of a safe shutdown earthquake. Eleven plants that received operating licenses prior to current seismic design criteria are presently being reevaluated as part of the Systematic Evaluation Program (SEP). Task A-40 was initiated to support this reevaluation and licensing activities in general by performing a short term review of seismic design criteria. This review will investigate alternative approaches to parts of the design sequence, quantify the overall conservatisms of the design sequence, and modify licensing criteria if changes are found to be justified.

Revision No. 2 of Task Action Plan A-40 states that operation of nuclear power plants can continue with reasonable assurance of protection to the health and safety of the public. This conclusion is based on the NRC staff's confidence in current design criteria since they anticipate the results of this task will confirm that current criteria provide an overall conservative approach to seismic design. While San Onofre Unit 1 was licensed before current design criteria, the plant is being seismically upgraded through the SEP based on these current criteria. As discussed below, necessary modifications are scheduled to be completed during the present outage. Since the seismic design of San Onofre Unit 1 is scheduled to be upgraded to current acceptance criteria before the plant returns to power operation, the generic basis for continued operation applies.

The NRC staff's letters dated January 1, 1980 and July 28, 1980 requested us to review the anchorages of safety related electrical equipment and non-seismic Category 1 auxiliary equipment at San Onofre Unit 1. This review identified anchorages of equipment in the plant and in the control room that required modifications. These modifications were completed and are documented in our letters dated May 23, 1980, March 25, 1981 and May 29, 1981.

By letter dated June 15, 1982 we advised the NRC staff that the structural upgrade program at San Onofre Unit 1 was being accelerated to accommodate a startup by November 28, 1982. Our letter dated June 24, 1982 provided the details of the implementation plan. The purpose of this program is to structurally upgrade San Onofre Unit 1 to withstand a 0.67g seismic event based on current criteria. This program specifically addresses: site ground motion, in situ soil conditions, structures, mechanical equipment and piping, electrical raceways and conduits, anchorage of electrical equipment, and the seismic backfit project (we also provided a report of the seismic backfit project by letter dated May 18, 1977).

The NRC staff has determined that continued operation of nuclear plants is warranted based on confidence in the conservatism of current licensing criteria. The above discussion outlines our program to upgrade the seismic

design of structures and equipment anchorages based on these current criteria. Since required modifications are scheduled to be completed during this outage, the generic basis applies to San Onofre Unit 1 and operation can continue with no undue risk to the health and safety of the public.

USI A-43

Containment Sump Emergency Performance

Task A-43 was initiated to address containment sump emergency performance. The emphasis of this program has been placed on nuclear plants licensed prior to 1974. Containment sumps in plants licensed after 1974 were evaluated according to current acceptance criteria established in Regulatory Guides 1.79 and 1.82. These current criteria are considered to establish conservative safety margins for sump performance. Under Task A-43, the ability of containment emergency sumps to maintain net positive suction head (NPSH) and resist flow blockage due to debris during emergency conditions will be evaluated. Acceptance criteria for plants licensed prior to 1974 will then be established based on the results of these evaluations.

Task Action Plan A-43 states that plants may continue to operate pending completion of the preliminary test program being conducted by Alden Research Laboratory. This decision is based on the favorable preliminary results of this test program. These results indicate that although a severe vortex may be formed for some configurations, the resulting void fraction in recirculation pipes is less than five percent. These conditions would cause only minimal pump flow degradation. Since these tests were conducted for a variety of containment sump configurations and the worst case produced only minimal flow degradation, individual plants can be expected to perform at least as well as the worst case. Therefore, the recirculation sump of San Onofre Unit 1 is expected to perform with only minimal flow degradation.

We have also taken steps to prevent sump suction blockage due to debris at San Onofre Unit 1. During 1974-1975, a seismic Category A baffle/screen arrangement was installed over the recirculation sumps. Sufficient screen area exists to prevent loss of adequate recirculation pump NPSH due to debris blockage. Our letter dated December 3, 1975 submits this modification as part of Amendment No. 52 to the Final Safety Analysis of San Onofre Unit 1.

We have shown that the two primary concerns of Unresolved Safety Issue A-43, assurance of adequate NPSH and resistance to flow blockage due to debris, are being addressed at San Onofre Unit 1. Assurance of adequate NPSH is based on tests conducted by Alden Research Laboratory. Resistance to flow blockage due to debris is provided by a seismic Category A baffle/screen. Since this issue is being adequately addressed at San Onofre Unit 1, operation can continue with no undue risk to the health and safety of the public.

USI A-44

STATION BLACKOUT

Task A-44 addresses the issue of station blackout at nuclear power plants. The complete loss of AC electrical power to the essential and non-essential switchgear busses in a nuclear power plant is referred to as a station blackout. Because many safety systems required for reactor core decay heat removal are dependent on AC power, the consequences of a station blackout could be a severe core damage accident. The purpose of Task A-44 is to evaluate the adequacy of current licensing design requirements to assure that nuclear power plants do not pose an unacceptable risk due to a station blackout incident.

The generic bases for continued operation are stated in Task Action Plan A-44. For an operating Westinghouse PWR such as San Onofre Unit 1, these bases are: (1) short term and long term AFW system requirements established by the Bulletins and Orders Task Force in response to the TMI-2 accident, and (2) a study of operating plants that did not identify any plants of unusually high susceptibility to a severe core damage accident resulting from a station blackout. We have reviewed these generic bases and determined they apply to San Onofre Unit 1 as discussed below.

The short term and long term requirements established by the Bulletins and Orders Task Force are compiled in NUREG-0645, "Report of the Bulletins and Orders Task Force." A complete discussion of the status of these requirements is presented in the section "USI A-45 - Shutdown Decay Heat Removal Requirements" of this report. Task Action Plan A-44 states that NUREG-0645 requirements of particular concern to USI A-44 are Recommendation GS-5 and Recommendation GL-1. Recommendation GS-5 requires plants be capable of providing the required AFW flow for at least 2 hours from one AFW pump train independent of any AC power source. Recommendation GL-1 requires a system to automatically initiate AFW flow. We have met these two requirements at San Onofre Unit 1. The NRC staff's Draft Safety Evaluation Report provided to us by letter dated December 7, 1981 indicated that our response to Recommendation GS-5 is adequate and that our response to GL-1 is under review.

As stated above, the second generic argument for continued operation is based on an NRC study of operating plants that did not identify any plants of unusually high susceptibility to a severe core damage accident resulting from a station blackout. We supported this study by providing the information requested by the NRC staff's letters dated December 15, 1977, September 25, 1979, February 25, 1981 (Generic Letter 81-04), and July 9, 1981. Except for Generic Letter 81-04, these letters only requested information on offsite and onsite AC power reliability and did not require specific actions. Our letters dated December 26, 1979 and January 18, 1980 submitted information on offsite AC power reliability and our letters dated January 19, 1978 and April 7, 1982 provided information on onsite AC power reliability.

Generic Letter 81-04 requested us to perform a review of San Onofre Unit 1 operating procedures and training programs to enhance our capability to mitigate a station blackout event. Our response dated July 21, 1981 indicated that a total loss of offsite and onsite AC power is unlikely at San Onofre Unit 1. This conclusion is based on the following: (1) offsite AC power is supplied by two independent sources-Southern California Edison Company and San Diego Gas and Electric Company, and (2) onsite AC power is supplied by two standby diesel generator trains, each capable of safely shutting down the reactor. However, our response indicated that we would implement the procedural and training recommendations contained in Generic Letter 81-04. The NRC staff's letter dated December 21, 1981 indicated that our responses to Generic Letter 81-04 are acceptable.

The continued operation of all Westinghouse PWRs is justified based on TMI-2 related AFW system requirements and a NRC study of operating plants that found no plant with an unusually high susceptibility to a severe core damage accident resulting from a station blackout. As discussed above, we have implemented the TMI-2 related requirements for the AFW system at San Onofre Unit 1. We also provided information to the NRC regarding AC power source reliability and implemented requirements contained in Generic Letter 81-04. San Onofre Unit 1 can therefore continue to operate with no undue risk to the health and safety of the public.

USI A-45

SHUTDOWN DECAY HEAT REMOVAL REQUIREMENTS

The overall purpose of Task A-45 is to evaluate the adequacy of current licensing design requirements in order to assure that nuclear power plants do not pose an unacceptable risk due to failure to remove shutdown decay heat. This requires the development of a comprehensive and consistent set of shutdown cooling requirements for existing and future plants, including the study of alternative means of shutdown decay heat removal and of diverse "dedicated" systems for this purpose. The outcome of Task A-45 will be the development of acceptance criteria for future plants and an evaluation of the acceptability of shutdown decay heat removal systems in existing plants.

Revision 1 of Task Action Plan A-45 states the generic bases for continued operation of all nuclear power plants pending resolution of this issue. The bases that apply to Westinghouse PWRs such as San Onofre Unit 1 include AFW system requirements and LOCA mitigation efforts resulting from the TMI-2 accident, and the ability of some plants to remove decay heat by the "feed and bleed" method in the event of an extended loss of feedwater.

NUREG-0645, "Report of the Bulletins and Orders Task Force," provides a compilation and status of TMI-2 related requirements concerning the AFW system and LOCA mitigation. These requirements were subsequently included in NUREG-0737, "The TMI Action Plan." The current status of AFW system requirements at San Onofre Unit 1 is documented in the NRC staff's Draft Safety Evaluation Report provided to us by letter dated December 7, 1981. The NRC staff's letter dated February 8, 1982 and our letter dated March 10, 1982 provide additional information regarding open items in the Draft SER. The only remaining open item for AFW requirements is GL-4, "Prevention of Multiple Pump Damage Due to Loss of Suction Resulting from Natural Phenomena." We have requested that modifications required to resolve this item be deferred until completion of the integrated assessment stage of the Systematic Evaluation Program (SEP). Our position is that modifications that may be required as a result of the SEP could impact the design details of the AFW system modifications.

The current status of LOCA mitigation requirements and other requirements contained in NUREG-0737 are provided in our letters dated April 16, 1982 and June 4, 1982. These letters also provide references that document our efforts to resolve all NUREG-0737 requirements. We have also installed circuitry to automatically trip the reactor coolant pumps in the event of a small break LOCA as recommended in NUREG-0623, "Generic Assessment of Delayed Reactor Coolant Pump Trip During Small Break Loss of Coolant Accidents in Pressurized Water Reactors."

In addition to our responses to the TMI-2 related requirements contained in NUREG-0645 and subsequently NUREG-0737, we have the capability to remove decay heat at San Onofre Unit 1 by the feed and bleed method in the event of an

extended loss of feedwater as documented in our May 20, 1982 letter. The ability to remove decay heat by the feed and bleed method fulfills one of the goals of Task Action Plan A-45 by providing an alternative means to remove decay heat using existing equipment if possible.

The bases for continued operation of Westinghouse PWRs stated in Task Action Plan A-45 include implementation of TMI-2 related requirements contained in NUREG-0645 and the ability to remove decay heat by the feed and bleed method. The above discussion provides information and identifies documentation that demonstrate our compliance with TMI-2 requirements. In addition, decay heat can be removed by the feed and bleed method at San Onofre Unit 1. We have therefore demonstrated that the generic bases for continued operation of Westinghouse supplied reactors apply specifically to San Onofre Unit 1 and continued operation will pose no undue risk to the health and safety of the public.

USI A-46

SEISMIC QUALIFICATION OF EQUIPMENT IN OPERATING PLANTS

Unresolved Safety Issue A-46 concerns seismic qualification of equipment in older operating nuclear plants. The mechanical and electrical equipment in these older nuclear power plants was designed before current acceptance criteria. To assure this equipment can bring the plant to a safe shutdown condition when subjected to a seismic event, the seismic qualification of this equipment must be reassessed. Instead of using current licensing criteria to qualify this equipment, more definitive criteria are needed to establish an acceptable safety margin. The primary activity of Task A-46 will be the development of criteria for use in assessing the capability of equipment important to safety to survive a seismic event and still perform its intended safety function.

The discussion of Task A-46 in NUREG-0705, "Identification of New Unresolved Safety Issues Related to Nuclear Power Plants," states that operating plants can continue to operate with no undue risk to the health and safety of the public, pending resolution of this issue. This conclusion is based on the NRC staff's experience gained through reviews of seismic issues, events and SEP facilities.

NUREG-0705 states that reviews of seismic issues and events have consisted of evaluating industrial facilities subjected to seismic events and evaluating design criteria of older nuclear plants. These reviews revealed that the design of the older nuclear plants followed rules and procedures that incorporated inherent conservatisms and used more rigorous techniques than other industrial facilities. Therefore, nuclear plants have large inherent safety margins. Since San Onofre Unit 1 was conservatively designed and constructed using rigorous techniques, we have concluded that the above discussed arguments apply to San Onofre Unit 1.

NUREG-0705 also states that when the design of SEP plants was considered on an integrated basis, equipment was adequate to resist the designated seismic hazard. Since San Onofre Unit 1 is one of the eleven SEP facilities under review, this argument applies to San Onofre Unit 1.

As part of the SEP review, many site specific modifications have been made to structurally upgrade San Onofre Unit 1 to withstand a ground motion of .67g. The anchors of safety related electrical equipment (Category 1) and non-seismic Category 1 auxiliary equipment were also evaluated and modified to assure the plant can be brought to safe shutdown conditions during a seismic event of up to 0.67g. The above SEP modifications and additional seismic information are discussed in this report in the section: "USI A-40 - Seismic Design Criteria - Short Term Program."

The above discussion demonstrates that the NRC staff's conclusion that nuclear plants can be safely operated until completion of Task A-46 applies to San Onofre Unit 1. We base this conclusion on the inherent safety margins involved in the design and construction of San Onofre Unit 1 and on modifications implemented as a result of the SEP. Therefore, San Onofre Unit 1 can continue to operate with no undue risk to the health and safety of the public.

USI A-47

Safety Implications of Control Systems

Task A-47 addresses the potential for transients or accidents being made more severe as a result of the failure or malfunction of control systems. This situation could occur if an adequate degree of separation and independence is not provided between safety grade and non-safety grade systems. The purpose of Task A-47 is to perform an evaluation of control systems and licensing criteria and propose necessary changes to assure nuclear power plants do not pose an unacceptable risk due to systems interactions.

Task Action Plan A-47 states the continued operation and licensing of PWRs and BWRs is acceptable pending completion of the Task A-47 program. The decision to allow continued operation is based on confidence in the adequacy of licensing review criteria and on actions required by Office of Inspection and Enforcement (IE) Information Notice 79-22, Bulletin 79-01, and Bulletin 79-27.

IE Information Notice 79-22 identified four control systems in which the failure of any one could exacerbate the effects of high energy line breaks. The notice further requested review of these postulated events to determine any necessary design or procedural modifications. Our October 5, 1979 letter submitted the results of our evaluation which concluded that our present design and procedures assure that the postulated events could be adequately mitigated. In addition, we indicated that since San Onofre Unit 1 is participating in SEP, high energy line breaks are also evaluated under SEP Topic III-5.A, "Effects of Pipe Break on Structures, Systems, and Components Inside Containment," and Topic III-5.B, "Pipe Break Outside Containment."

IE Bulletin 79-01, "Environmental Qualification of Class 1E Equipment," notified licensees of requirements to assure adequate qualification of equipment exposed to high energy line break environments. However, plants participating in SEP were exempted from actions required by this bulletin. Therefore, no response was required for San Onofre Unit 1. A complete discussion of our efforts concerning qualification of safety related equipment is presented in the section "USI A-24 - Qualification of Class 1E Safety Related Equipment" of this submittal.

IE Bulletin 79-27, "Loss of Non-Class 1E Instrumentation and Control Power System Bus During Operation," requested that evaluations of plant procedures and designs be performed to ensure the adequacy of the plant to accomplish shutdown upon loss of power to any electrical bus supplying power for instruments and controls. Our response was submitted by letter dated May 15, 1980 which identified several design and procedural modifications to be implemented according to the schedule contained in our letter. These modifications increase the independence and diversity of control systems, provide the operators with additional control system status indicators, and provide procedures to assure safe shutdown.

As part of SEP, we are taking actions which relate to Unresolved Safety Issue A-47 in addition to those actions discussed above. These actions are under SEP Topic VII-1.A, "Isolation of the Reactor Protection System from Non-Safety Systems, Including Qualification of Isolation Devices," Topic VII-2, "Engineering Safety Features (ESF) System Control Logic and Design," and Topic IV-2, "Reactivity Control Systems." Resolution of requirements identified in the integrated assessment will assure an adequate degree of separation and independence is provided between non-safety grade systems and safety grade systems.

The NRC staff has concluded that continued operation of reactors is acceptable based on licensing criteria and requirements identified in Task Action Plan A-47. We reviewed these bases and in the above discussion demonstrated that we have taken necessary action in order for these bases to apply to San Onofre Unit 1. In addition, we described SEP topics that specifically address the safety implications of control systems to provide additional assurance that Unresolved Safety Issue A-47 is being adequately addressed at San Onofre Unit 1. Therefore, San Onofre Unit 1 can continue to operate with no undue risk to the health and safety of the public.

USI A-48

HYDROGEN CONTROL MEASURES AND EFFECTS OF HYDROGEN BURNS ON SAFETY EQUIPMENT

Unresolved Safety Issue A-48 concerns the potential unacceptable generation of hydrogen following a reactor accident. Before the Three Mile Island Unit 2 (TMI-2) accident, licensing criteria were based on 10 CFR 50.44, "Standards for Combustible Gas Control Systems in Light Water Cooled Power Reactors." However, the accident at TMI-2 resulted in a metal-water reaction which involved hydrogen generation well in excess of the design basis amounts specified in 10 CFR 50.44. As a result it became apparent that additional hydrogen control measures would have to be taken at all nuclear power plants. Task A-48 was initiated to assure integration of various NRC efforts into a unified program with a generic resolution.

Task Action Plan A-48 separates containments into the following categories: (1) Small Volume Mark I and II Pressure Suppression containments, (2) Intermediate Volume Ice Condenser and Mark III Pressure Suppression containments, and (3) Large Volume Dry containments. The task action plan further concludes that continued operation and licensing of plants in each of these three categories is warranted based on category specific reasons.

The containment structure of San Onofre Unit 1 has a free volume of 1.2 million cubic feet and a design pressure of 46 psig. Task Action Plan A-48 Category 3 containments have a representative free volume of two million cubic feet and design pressures from 45 to 60 psi. The free volume difference between the containment at San Onofre Unit 1 and Category 3 containments can be attributed to the lower power rating of San Onofre Unit 1 (1,347 MW thermal) compared to Category 3 reactors (some are rated at over 3,000 MW thermal). Therefore, the containment of San Onofre Unit 1 is a Category 3 containment and the bases stated in Task Action Plan A-48 for Category 3 containments apply to San Onofre Unit 1. These bases consider the large free volume available to dilute the hydrogen and the ability of the high design pressure containment to withstand hydrogen combustion.

In addition to the generic bases for continued operation stated in Task Action Plan A-48 and discussed above, site specific actions were taken at San Onofre Unit 1 as a result of the TMI-2 accident. These actions were taken in accordance with NUREG-0578, "Short Term Lessons Learned." Two studies were conducted at San Onofre Unit 1 to determine (1) post loss of coolant accident hydrogen concentrations in containments and (2) the post loss of coolant accident mixing of combustible gases. The first study revealed unacceptable hydrogen concentrations. We responded to this study and to NUREG-0578 Section 2.1.5.a by installing redundant safety grade hydrogen recombiners inside containment. This modification is documented in our letters dated October 17, 1979 and January 17, 1980. The second study indicated that the

containment spray system not only served as an excellent containment gas blending device, but also effectively prevented hydrogen combustion by maintaining a high steam volume inside containment. Therefore, no modifications were necessary to enhance combustible gas mixing inside containment.

We have reviewed the bases for continued operation contained in Task Action Plan A-48 and found they apply to San Onofre Unit 1. Also, in response to NUREG-0578 Section 2.1.5.a, redundant safety grade hydrogen recombiners were installed inside containment as discussed above. In addition, a study indicated that the containment spray system will effectively mix gases inside containment and prevent combustion. Therefore, San Onofre Unit 1 can continue to operate with no undue risk to the health and safety of the public.

USI A-49

PRESSURIZED THERMAL SHOCK (PTS)

Task A-49 was initiated to address potential severe PWR pressure vessel overcooling events which could be followed by repressurization of the reactor vessel (pressurized thermal shock events). Rapid overcooling of the pressure vessel causes a temperature distribution across the vessel wall which results in thermal stress that is further compounded if the vessel is repressurized. Postulated causes of a potential PTS event include instrumentation and control system malfunctions, small break loss of coolant accidents, main steamline breaks, and feedwater pipe breaks. The purpose of Task A-49 is to formulate regulatory requirements to ensure that the risk of pressure vessel failure from PTS events is sufficiently low through each vessel's design life.

Task Action Plan A-49 states that nuclear power plants can continue to operate pending resolution of the pressurized thermal shock issue. This conclusion is based on the low probability of a reactor pressure vessel failure. In order for a reactor pressure vessel to fail, an unlikely combination of factors must occur. These factors are: (1) a reactor vessel flaw of sufficient size to initiate and propagate; (2) a level of irradiation and properties and composition sufficient to cause significant embrittlement of the material (present irradiation levels and RT_{NDT} at operating reactors are sufficiently low to provide significant margin to failure); (3) a severe overcooling transient with repressurization (you have estimated that the probability of a severe overcooling transient in a Westinghouse designed plant is lower than 10^{-3} per reactor year by as much as an order of magnitude); and (4) the crack resulting from the initial crack's propagation must be of such size and location that the vessel fails. We have reviewed this reasoning for determining that a PTS induced failure is a low probability event and determined that the same unlikely combination of factors must also occur at San Onofre Unit 1 in order for a PTS induced failure to occur. Therefore, the probability of a pressure vessel failure at San Onofre Unit 1 is acceptably low.

On a plant specific basis, San Onofre Unit 1 was included as one of the eight lead plants requested to respond to the NRC's August 21, 1981 letter. This letter requested information be provided in 60 and 150 day responses, which were provided to the NRC by letters dated November 4, 1981 and January 25, 1982. Our 150 day response indicated that the San Onofre Unit 1 reactor vessel was acceptable from a PTS standpoint until the end of its design life. Subsequently, we provided an evaluation of San Onofre Unit 1 by letter dated May 20, 1982 in response to the NRC staff's letter dated April 7, 1982. This evaluation formed the basis for the emergency procedures concerning PTS which were submitted to the NRC staff by letter dated September 13, 1982. In addition to these actions, we have been an active participant in the Westinghouse Owners Group which is working on the generic resolution of this issue. In addition, the NRC staff has issued their draft evaluation which indicates the San Onofre Unit 1 reactor vessel is acceptable for its design life.

The NRC staff has determined that the probability of a reactor pressure vessel failure is sufficiently low to justify continued operation pending resolution of the PTS issue. The above discussion indicates that we consider this generic argument for continued operation to apply to San Onofre Unit 1. We also discuss above site specific assurance that San Onofre Unit 1 will not experience a PTS induced pressure vessel failure. Therefore, San Onofre Unit 1 can continue to operate with no undue risk to the health and safety of the public.

GLR:5894