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Case No.: 2016-0061
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FOIA Resource

From: Mike Aguirre <maguirre@amslawyers.com>
Sent: Monday, October 26, 2015 2:50 PM
To: FOIA Resource
Subject: [External_Sender] FW: Starting Point for Investigation "letters, e-mails, meeting minutes, action item lists, and internal memoranda"
Attachments: findings.pdf; ML121350603-3.pdf; summary of meeting.pdf; NRC Releases Missing Record on San Onofre Plant Design Changes _ Voice of OC.pdf

Greetings: Please provide to me under the U.S. Freedom of Information Act (FOIA) the "letters, e-mails, meeting minutes, action item lists, and internal memoranda that suggest[ed] concerns" as stated on page 25 of the NRC attached 20 September 2013 findings Re: SUBJECT: SAN ONOFRE NUCLEAR GENERATING STATION – NRC CONFIRMATORY ACTION LETTER RESPONSE INSPECTION 05000361/2012009 AND 05000362/2012009.

The pertinent part of the findings states:

> and resultant design. From shortly after the contract was awarded until 2006, there were letters, e-mails, meeting minutes, action item lists, and internal memoranda that suggested concerns with all three of the elements that cause fluid-elastic instability, which is void fraction, gap velocity, and adequacy of anti-vibration bar tube supports. Regarding concerns raised about FIT-III gap velocities, Mitsubishi compared the velocities to other Mitsubishi designed triangular pitch steam generators that also used FIT-III, but did not compare the results to other similar-sized steam generators.

Also the NRC in May 2012 released the attached Edison power point purportedly presented to the NRC on 7 June 2006 at the NRC office in Maryland. However, the NRC indicated in response to the Friends of the Earth (A. Gunderson) FOIA that the NRC did not have a copy of the 7 June 2006 Edison power point handout for the 7 June 2006 meeting 7 June 2006 handout) (attached). The NRC indicated it would have to request a copy of the Edison power point hand out from the 7 June 2006 meeting. (See attached article)

Please provide all records of communication showing how and when any NRC agent, officer or employee requested and obtained a copy of the 7 June 2006 handout from Edison. Also please provide any record or writing explaining how and why the NRC did not have a copy of the 7 June 2006 handout.
I will pay for the copy costs of the documents requested under this FOIA request. Time is of the essence and I do not agree to any continuances. If the NRC fails to provide these requested documents litigation will be filed under the strict guidelines of the FOIA.

Michael J. Aguirre
Aguirre & Severson

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Michael J. Aguirre
Aguirre & Severson

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
1600 EAST LAMAR BLVD
ARLINGTON, TEXAS 76011-4511

September 20, 2013

CAL 4-12-001
EA-13-083

Mr. Peter Dietrich
Senior Vice President and
Chief Nuclear Officer
Southern California Edison Company
San Onofre Nuclear Generating Station
P.O. Box 128
San Clemente, CA 92674-0128

EA-13-083

SUBJECT: SAN ONOFRE NUCLEAR GENERATING STATION – NRC CONFIRMATORY ACTION LETTER RESPONSE INSPECTION 05000361/2012009 AND 05000362/2012009

Dear Mr. Dietrich:

Following the June 7, 2013, announcement of Southern California Edison's decision to permanently shut down San Onofre Nuclear Generating Station, Units 2 and 3, the U.S. Nuclear Regulatory Commission (NRC) terminated our review of your Confirmatory Action Letter Response (ML12285A263) for Unit 2, dated October 3, 2012. The enclosed report documents the NRC assessment of your activities through June 7, 2013, in response to our March 27, 2012, Confirmatory Action Letter (ML12087A323). The NRC also reviewed the two remaining open unresolved items identified in Augmented Inspection Team Report 05000361/2012007 and 05000362/2012007 (ML12188A748). The two unresolved items were related to the mechanistic cause of the excessive and unexpected wear in both Units 2 and 3 steam generator tubes, which resulted in a steam generator tube leak on Unit 3 on January 31, 2012. The results of this inspection were discussed with you and other members of your staff on August 28, 2013.

The inspectors examined activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license. The inspectors reviewed selected procedures, documents, and records and interviewed personnel.

On June 12, 2013, Southern California Edison submitted a Certification of Permanent Cessation of Power Operations letter to the NRC, certifying that Units 2 and 3 have permanently ceased power operations. On June 28 and July 22, 2013, Southern California Edison certified that all fuel had been permanently removed from the Units 3 and 2 reactors, respectively (ML13183A391 and ML13204A304).

Additionally, the inspectors reviewed the replacement steam generator design specification for the replacement steam generators to identify the applicable design standards for thermal-hydraulic modeling and flow-induced vibration. The review of design information included design basis documents for the original steam generators to identify any design requirements for thermal-hydraulic modeling and flow-induced vibration, in order to determine if those requirements were properly translated into the replacement steam generator design specification. The inspectors reviewed the applicable design standards to identify design information that would have prompted the licensee to identify deficiencies in the thermal-hydraulic model and flow-induced vibration analysis. Particularly, the inspectors reviewed the technical justification for critical assumptions and design inputs.

The inspectors interviewed licensee staff and reviewed applicable quality assurance requirements and site procedures for the verification of supplier documents to assess whether the licensee had a reasonable opportunity to identify any deficiencies with the thermal-hydraulic modeling and the flow induced-vibration analysis based on the requirements and guidance in site procedures.

b. Findings

Introduction. For Unit 2, the NRC identified a Green noncited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," for the failure to verify the adequacy of the thermal-hydraulic and flow-induced vibration design of the replacement steam generators.

For Unit 3, the NRC identified an apparent violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," for the failure to verify the adequacy of the thermal-hydraulic and flow-induced vibration design of the replacement steam generators, which also resulted in an associated apparent violation of Technical Specification 5.5.2.11, "Steam Generator Program," because of a loss of tube integrity on Unit 3 Steam Generator 3E0-88.

Description. The effective construction code for SCE replacement steam generators was the 1998 Edition, with 2000 Addenda, of the ASME BPVC, Section III. Article NCA-3200 required that the owner shall prepare, review, and approve the replacement steam generator design specification, which contains the specific design requirements for the applicable code component. The replacement steam generator design specification for SCE replacement steam generators (Document SO23-617-01, "Specification for Design and Fabrication of replacement steam generators for Unit 2 and Unit 3," Revision 4), Section 3.8.2, stated that: "The Supplier [MHI] shall prepare and submit for SCE's approval a Performance Analysis Report documenting all thermal-hydraulic aspects of the replacement steam generators. The Report shall include all computer codes and modeling for the thermal-hydraulic performance of the replacement steam generators. The Report shall include detailed calculations, by region, showing that cross-flow velocities within the tube bundle shall be such as to

minimize tube wear at the tube to tube-support interfaces. The calculations shall clearly identify the damping factor(s) used and margins to flow instability for steam flow rates of up to 120% of the design flow rate."

Additionally, Section 3.21.7 of the replacement steam generator design specification stated that, "The Supplier shall provide a new thermal-hydraulic analytical model, or update the existing plant EPRI ATHOS model for the replacement steam generators, and furnish all input parameters required to update the existing steam generator simulator model. The Supplier shall provide an executable version of the thermal-hydraulic computer codes used in the design of the replacement steam generators."

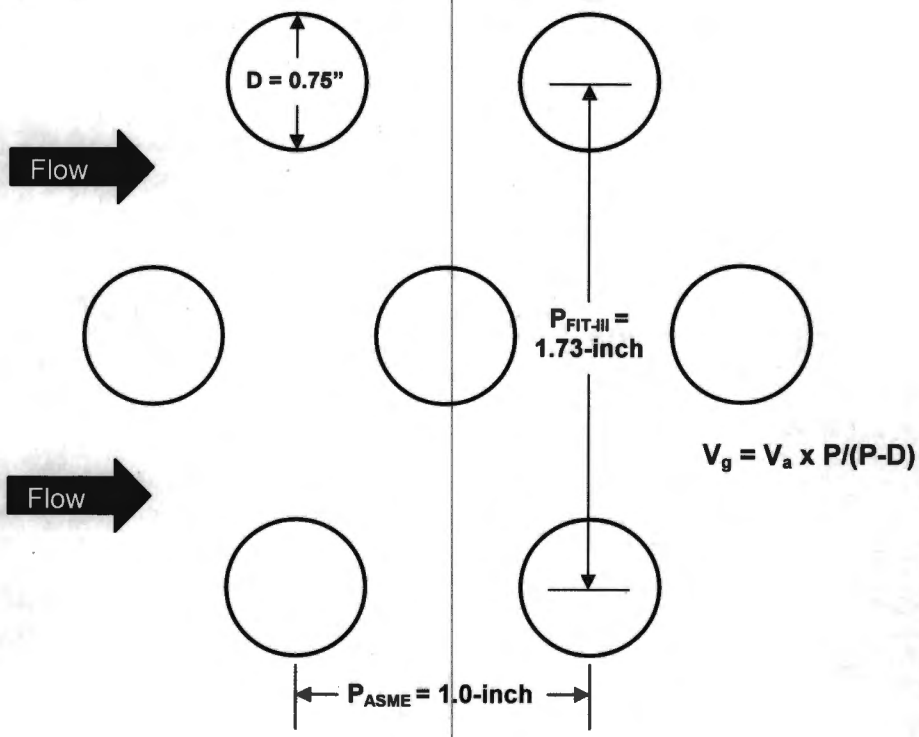
Furthermore, Section 3.5.1 of the replacement steam generator design specification stated that, "To the extent practical, the version and identity of all Codes, Standards, and other documents applicable to this Specification are shown in this Section." Following this statement, the replacement steam generator design specification listed ASME BPVC, Section III, Subsection NCA, and Division 1 Appendices as part of the applicable standards.

Mitsubishi Document L5-04GA504, "Evaluation of Tube Vibration," Revision 3, adopted the methodology in Non-Mandatory Appendix N to ASME BPVC, Section III, "Dynamic Analysis Methods," to evaluate flow-induced vibration in tube arrays exposed to cross flow. Specifically, Section 5 of this document stated that the vibration analysis was performed in accordance with the procedures and suggested inputs given in Appendix N-1330 to ASME Code Section III. This analysis was reviewed and approved by SCE on January 28, 2008, during the design stage of the replacement steam generators.

Paragraph N-1330 in Appendix N, to ASME Code Section III, provided recommendations and inputs for avoiding fluid-elastic instability of tube arrays. Paragraph N-1331.1, "Prediction of the Critical Velocity," stated that the onset of instability is governed, in part, by the flow velocity in the gaps between the tubes, which is determined by $Vg = Va \times P/(P-D)$, where "Vg" is the gap velocity, "Va" is the approach flow velocity that would occur if the tubes were not present, "P" is the tube array pitch as defined in Figure N-1331-3, and "D" is the outside diameter of a tube.

In response to this unresolved item, Mitsubishi identified that one of the factors responsible for the nonconservative flow velocities was that the flow area definition was not consistent with the recommendations in Appendix N (Mitsubishi Document L5-04GA591, "Validity of Use of the FIT-III Results during Design," Revision 1). Mitsubishi determined that the tube-to-tube gap used in the FIT-III thermal-hydraulic code, to determine the gap velocities, was larger than the recommended value in Appendix N, which resulted in lower calculated flow velocities. The difference in flow area definition is illustrated below. The tube array in the SCE replacement steam generators is a triangular array rotated 60 degrees, with a tube pitch of P = 1.0-inch. For that type of array, ASME BPVC, Section III, Appendix N, Figure N-1331-3, defines the tube pitch as the center-to-center distance between two tubes along the same column/row and in the longitudinal direction of the flow, which in this case would be

P = 1.0 inch. However, the flow area defined in FIT-III used the tube pitch in the transverse direction of the flow, which in SCE-rotated triangular array would be P = 1.73 inches. The use of a larger pitch in the FIT-III thermal-hydraulic analysis resulted in nonconservative calculated (lower) flow velocities.



Mitsubishi Document L5-04GA521, "Three-Dimensional Thermal and Hydraulic Analysis," Revision 3, performed by Mitsubishi and approved by SCE during the design phase of the replacement steam generators, showed that the thermal-hydraulic model was built with two different pitch values. The report stated that the model was built with a 1.0-inch pitch in the longitudinal direction of flow and 1.73-inch pitch in the transverse direction. This analysis report was approved by SCE on April 2, 2008. Additionally, the "Evaluation of Tube Vibration" report by Mitsubishi stated that the thermal-hydraulic conditions for the FIT-III modeling were based on a 1.0-inch pitch in the longitudinal direction of flow and 1.73-inch pitch in the transverse direction. These two design calculations were supporting documents for the "Performance Analysis Report" required in the replacement steam generator design specification. As indicated above, FIT-III's output for gap velocity results used the 1.73-inch distance instead of the 1.0-inch distance. The FIT-III code was developed by Takasago, MHI's research and development center. Takasogo was responsible for conducting the thermal-hydraulic analysis, using FIT-III, for each steam generator design. The FIT-III results were then provided to the MHI Steam Generator Design

Department, which input the gap velocity information into the FIVATS vibration code. However, it was not recognized that the gap velocities input into the vibration code were incorrect.

The inspectors determined that the licensee did not ensure that the thermal-hydraulic modeling and flow-induced vibration analysis of the replacement steam generators were adequate with respect to the replacement steam generator design specification. Specifically, the licensee failed to ensure that the design calculations appropriately incorporated the methodology from the ASME BPVC, Section III, Appendix N, standard that was adopted by Mitsubishi for the flow-induced vibration analysis. There were opportunities to identify this error during the early design stage of the replacement steam generators. Licensee personnel questioned the analysis results of FIT-III during design review meetings, but ultimately accepted the model results and resultant design. From shortly after the contract was awarded until 2006, there were letters, e-mails, meeting minutes, action item lists, and internal memoranda that suggested concerns with all three of the elements that cause fluid-elastic instability, which is void fraction, gap velocity, and adequacy of anti-vibration bar tube supports. Regarding concerns raised about FIT-III gap velocities, Mitsubishi compared the velocities to other Mitsubishi designed triangular pitch steam generators that also used FIT-III, but did not compare the results to other similar-sized steam generators.

As a result of the failure to verify the adequacy of the thermal-hydraulic and flow-induced vibration design, both Unit 3 replacement steam generators experienced fluid-elastic instability in a localized area of the tube bundle leading to rapid, significant, unexpected tube-to-tube wear. The tube degradation progressed to the point of causing a primary-to-secondary leak in Steam Generator 3E0-88 through Tube R106C78. Additionally, from March 13-21, 2012, the licensee conducted in-situ pressure testing of the suspect tubes in both Unit 3 steam generators and identified a total of eight tubes (including the leaking tube) that failed to meet the performance criteria in plant Technical Specifications. The in-situ pressure testing identified that Tubes R106C78, R102C78, R104C78, R100C80, R107C77, R101C81, R98C80, and R99C81 in Steam Generator 3E0-88 failed to meet the structural integrity criterion in Technical Specification 5.5.2.11. In addition to failing the structural integrity criterion, Tubes R106C78, R102C78, and R104C78 also failed to meet the accident-induced leakage criterion in Technical Specification 5.5.2.11.

Southern California Edison completed a review of the tube failures, including conducting a deterministic root cause, an organization and programmatic root cause (still ongoing), three different operational assessments, modification testing, and submittal of a response dated October 3, 2012 (ML12285A263) to the NRC's March 27, 2012, Confirmatory Action Letter (ML 12087A323). The organizational and programmatic root cause evaluation has not been completed as of the issuance of this report, in order to identify the causes of the breakdown in design control such that comprehensive corrective actions can be taken to not only prevent recurrence, but prevent the failures of other important structures, systems, and components that may be subject to the same or similar design problems.

Unit 2:

Analysis. The inspectors determined that the licensee's failure to verify the adequacy of the thermal-hydraulic and flow-induced vibration design of the replacement steam generators was a performance deficiency. Criterion III specifies that design control measures shall provide for verifying or checking the adequacy of design, in particular, thermal and hydraulic analyses. This performance deficiency is more than minor, and therefore a finding, because it is associated with the equipment performance attribute of the Initiating Event Cornerstone and adversely affected the cornerstone objective of limiting the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations.

The inspectors used NRC Inspection Manual Chapter 0609, Attachment 4 and Appendix A, to evaluate the significance of this finding. In accordance with Exhibit 1 of Inspection Manual Chapter 0609, Appendix A, the inspectors determined that the finding is of very low safety significance because the finding did not involve a degraded steam generator tube that could not sustain three times the normal operating differential pressure and did not violate the accident leakage performance criterion.

The licensee initiated Nuclear Notification NN 202447268 to address this issue in the corrective action program and implement corrective actions to prevent recurrence. Southern California Edison revised the thermal-hydraulic code of record and ensured that the code was in accordance with ASME guidance.

No crosscutting aspect was assigned because this performance deficiency occurred in the 2005 to 2008 timeframe. Substantial management and personnel changes have occurred, including taking actions to address a chilled work environment and various crosscutting themes. The NRC determined that the performance behavior that existed at that time is not indicative of current performance.

Enforcement. Title 10 CFR Part 50, Appendix B, Criterion III, "Design Control," requires, in part, that design control measures shall be established to provide for verifying or checking the adequacy of design, such as by the performance of design reviews, by the use of alternate or simplified calculational methods or by the performance of a suitable testing program.

Contrary to the above, on January 28, 2008, and April 2, 2008, SCE failed to verify or check the adequacy of Mitsubishi's developed design Documents L5-04GA504 (SO23-617-1-C157), "Evaluation of Tube Vibration," Revision 3, and L5-04GA521 (SO23-617-1-C683), "Three-Dimensional Thermal and Hydraulic Analysis," Revision 3, respectively, for the flow-induced vibration and thermal-hydraulic designs. Specifically, the output of the thermal-hydraulic code and input to the vibration code were not verified or checked to be in accordance with ASME Section III, Appendix N, "Dynamic Analysis Methods." Because the finding is of very low safety significance and has been entered into the licensee's corrective action program as Nuclear Notification NN 202447268, this violation is being treated as a noncited violation

consistent with Section 2.3.2 of the NRC Enforcement Policy; NCV 05000361/2012009-01, "Failure to Verify Adequacy of Thermal-Hydraulic and Flow-Induced Vibration Design for the Unit 2 Replacement Steam Generators."

Unit 3:

Analysis. Regarding Unit 3, this failure also constitutes a performance deficiency for the same reason previously discussed for Unit 2. Specifically, the failure to verify the adequacy of the thermal-hydraulic and flow-induced vibration design resulted in significant and unexpected steam generator tube wear due to fluid-elastic instability, which challenged the structural integrity of the steam generator tubes to perform their pressure boundary function.

The inspectors used NRC Inspection Manual Chapter 0609, Attachment 4 and Appendix A, to evaluate the significance of this finding. In accordance with Exhibit 1 of Inspection Manual Chapter 0609, Appendix A, the inspectors determined that this finding required evaluation in accordance with Inspection Manual Chapter 0609, Appendix J, because the finding involved a degraded steam generator tube condition, where one tube cannot sustain three times the differential pressure across a tube during normal full power, steady-state operation. In accordance with Inspection Manual Chapter 0609, Appendix J, this finding required a detailed risk analysis, since it involved two or more tubes that could not sustain three times the normal differential pressure and one or more steam generators that violated "accident-induced leakage" performance criterion. A Phase 3 analysis was completed using the San Onofre SPAR model, Revision 8.22, assuming average test and maintenance, and a truncation limit of 1.0E-11. Based on the best available information, the performance deficiency was preliminarily characterized as a finding of low to moderate safety significance (White). Refer to Attachment 4 for the detailed Phase 3 analysis.

The licensee initiated Nuclear Notification NN 202447265 to address this issue in the corrective action program. Southern California Edison revised the thermal-hydraulic code of record and ensured that the code was in accordance with ASME guidance.

No crosscutting aspect was assigned because this performance deficiency occurred in the 2005 to 2008 timeframe. Substantial management and personnel changes have occurred, including taking actions to address a chilled work environment and other safety culture issues. The NRC determined that the performance behavior that existed at that time is not indicative of current performance.

Enforcement. Title 10 CFR Part 50, Appendix B, Criterion III, "Design Control," states, in part, that design control measures shall be established to provide for verifying or checking the adequacy of design, such as by the performance of design reviews, by the use of alternate or simplified calculational methods or by the performance of a suitable testing program.

Technical Specification 5.5.2.11, "Steam Generator Program," Section b, "Performance criteria for SG tube integrity," states, in part, that steam generator tube

integrity shall be maintained by meeting the performance criteria for tube structural integrity and accident induced leakage. Technical Specification 5.5.2.11 b.1, "Structural integrity performance criterion," states, in part, that this includes retaining a safety factor of 3.0 against burst under normal steady-state full power primary-to-secondary differential pressure. Technical Specification 5.5.2.11 b.2, "Accident induced leakage performance criterion," states, in part, that leakage shall not exceed 0.5 gallons per minute per steam generator for a main steam line break accident.

Contrary to the above, on January 28 and April 2, 2008, SCE failed to verify or check the adequacy of Mitsubishi's developed design Documents L5-04GA504 (SO23-617-1-C157), "Evaluation of Tube Vibration," Revision 3, and L5-04GA521 (SO23-617-1-C683), "Three-Dimensional Thermal and Hydraulic Analysis," Revision 3, respectively, for the flow-induced vibration and thermal-hydraulic designs. Specifically, the output of the thermal-hydraulic code and input to the vibration code were not verified or checked to be in accordance with ASME Section III, Appendix N, "Dynamic Analysis Methods."

Consequently, the inadequate thermal-hydraulic and flow-induced vibration design resulted in adverse flow conditions, along with insufficient tube support, which caused fluid-elastic instability of a group of tubes in both Unit 3 replacement steam generators. This resulted in one tube leaking and required operator response to rapidly shut down Unit 3 on January 31, 2012. In March 2012, in-situ pressure testing on Unit 3 Steam Generator 3E0-88 confirmed that eight steam generator tubes failed to meet the performance criterion for structural integrity and three of those tubes also failed to meet the accident-induced leakage criterion. During in-situ pressure testing, Tubes R106C78, R102C78, R104C78, R100C80, R107C77, R101C81, R98C80, and R99C81 in Steam Generator 3E0-88 failed to meet the structural integrity criterion limit of three times the normal steady-state primary-to-secondary differential pressure of 5250 psig (room temperature equivalent to 4290 psi under hot 100 percent power conditions), with the tubes failing at test pressures ranging from 2874 psig to 5026 psig (at room temperature). In addition, Tubes R106C78, R102C78, and R104C78 failed to meet the accident-induced leakage criterion of not exceeding 0.5 gpm leakage per steam generator at a main steam line break test pressure of 3200 psig (room temperature equivalent to 2560 psig differential pressure during main steam line break), with each tube having leakage rates of approximately 4.5 gpm, prior to exceeding 3200 psig. Because this finding has been preliminarily determined to be of low-to-moderate safety significance (White), it will be treated as an apparent violation and tracked as AV 05000362/2012009-02, "Failure to Verify Adequacy of Thermal-Hydraulic and Flow-Induced Vibration Design for the Unit 3 Replacement Steam Generators."

June 16, 2006

LICENSEE: SOUTHERN CALIFORNIA EDISON (SCE)

FACILITIES: SAN ONOFRE NUCLEAR GENERATING STATION, UNITS 2 AND 3 (SONGS 2 AND 3)

SUBJECT: MEETING WITH REPRESENTATIVES OF SCE FOR SONGS 2 AND 3

A meeting was held on Wednesday, June 7, 2006, between the Nuclear Regulatory Commission (NRC) staff and the SCE, the licensee for SONGS 2 and 3. The meeting was held at the request of the licensee to provide to the NRC staff an overview of the various aspects of its steam generator (SG) replacement project. The notice for the meeting was issued on May 25, 2006.

The replacement schedule for Unit 2 is the 4th quarter of 2009 and for Unit 3, it is the 4th quarter of 2010, during outage cycle number 16 for both units. The SGs are being fabricated by Mitsubishi Heavy Industries in Kobe, Japan. The presentation also included a schedule of various activities, including fabrication, engineering, and installation; current status, licensing issues, transportation, and implementation disposal of the original SGs; site/space limitations, and containment access. Representatives from the Containment and Ventilation Branch, SG Tube Integrity and Chemical Engineering Branch, PWR [Pressurized-Water Reactor] Systems Branch, Engineering Mechanics Branch, and Operating Reactors Licensing Branch attended the presentation.

Enclosure 1 is the list of attendees. Enclosure 2 is the material handed out by the licensee. There was no handout from the NRC staff.

/RA/

N. Kalyanam, Project Manager
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos.: 50-361 and 50-362

Enclosures: 1. List of Attendees
2. Licensee's Handout

cc w/encls: See next page

June 16, 2006

LICENSEE: SOUTHERN CALIFORNIA EDISON (SCE)

FACILITIES: SAN ONOFRE NUCLEAR GENERATING STATION, UNITS 2 AND 3 (SONGS 2 AND 3)

SUBJECT: MEETING WITH REPRESENTATIVES OF SCE FOR SONGS 2 AND 3

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N. Kalyanam, Project Manager
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos.: 50-361 and 50-362

Enclosures: 1. List of Attendees
2. Licensee's Handout

cc w/encls: See next page

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LIST OF ATTENDEES AT THE JUNE 7, 2006, MEETING

NAME	AFFILIATION
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D. Spiker	SCE
J. Rainsberry	SCE
L. Pressey	SCE
M. Wharton	SCE
D. Terao	NRR/DORL/LPLIV
G. Tesfaye	NRR/DSS/SCVB
J. Nakoski	NRR/DSS/SPWB
G. Makar	NRR/DCI/CSGB
A. Lee	NRR/ADES/EEMB
N. Kalyanam	NRR/DORL/LPLIV

Where: SCE = Southern California Edison
NRR = Office of Nuclear Reactor Regulation

ENCLOSURE 1

LICENSEE'S HANDOUT FOR JUNE 7, 2006, MEETING

SAN ONOFRE NUCLEAR GENERATING STATION, UNITS 2 AND 3

STEAM GENERATOR REPLACEMENT

PROJECT OVERVIEW

ENCLOSURE 2

San Onofre Nuclear Generating Station
Units 2 and 3

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Mr. Ray Waldo, Vice President
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Mr. Brian Katz
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February 2006

cc:
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Department of Health Services
Radiologic Health Branch
MS 7610, P.O. Box 997414
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February 2006

SOUTHERN CALIFORNIA EDISON

An EDISON INTERNATIONAL® Company

San Onofre Nuclear Generating Station Units 2 & 3

STEAM GENERATOR REPLACEMENT PROJECT OVERVIEW

June 7, 2006

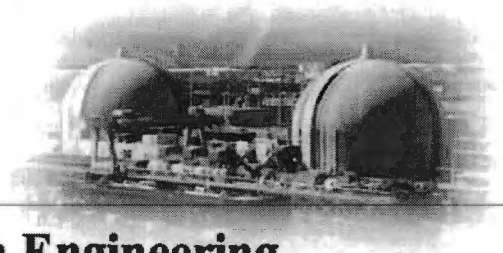
SONGS SGR TEAM

Agenda

- Background
- Schedule
- Current Status
- Licensing
- Replacement Steam Generators
- Transportation
- Implementation
- Disposal of Original Steam Generators
- Q&A

SONGS Background

- Started SONGS 2 & 3 Construction: 1974
- Commercial Operation: August 1983 - Unit 2
  April 1984 - Unit 3

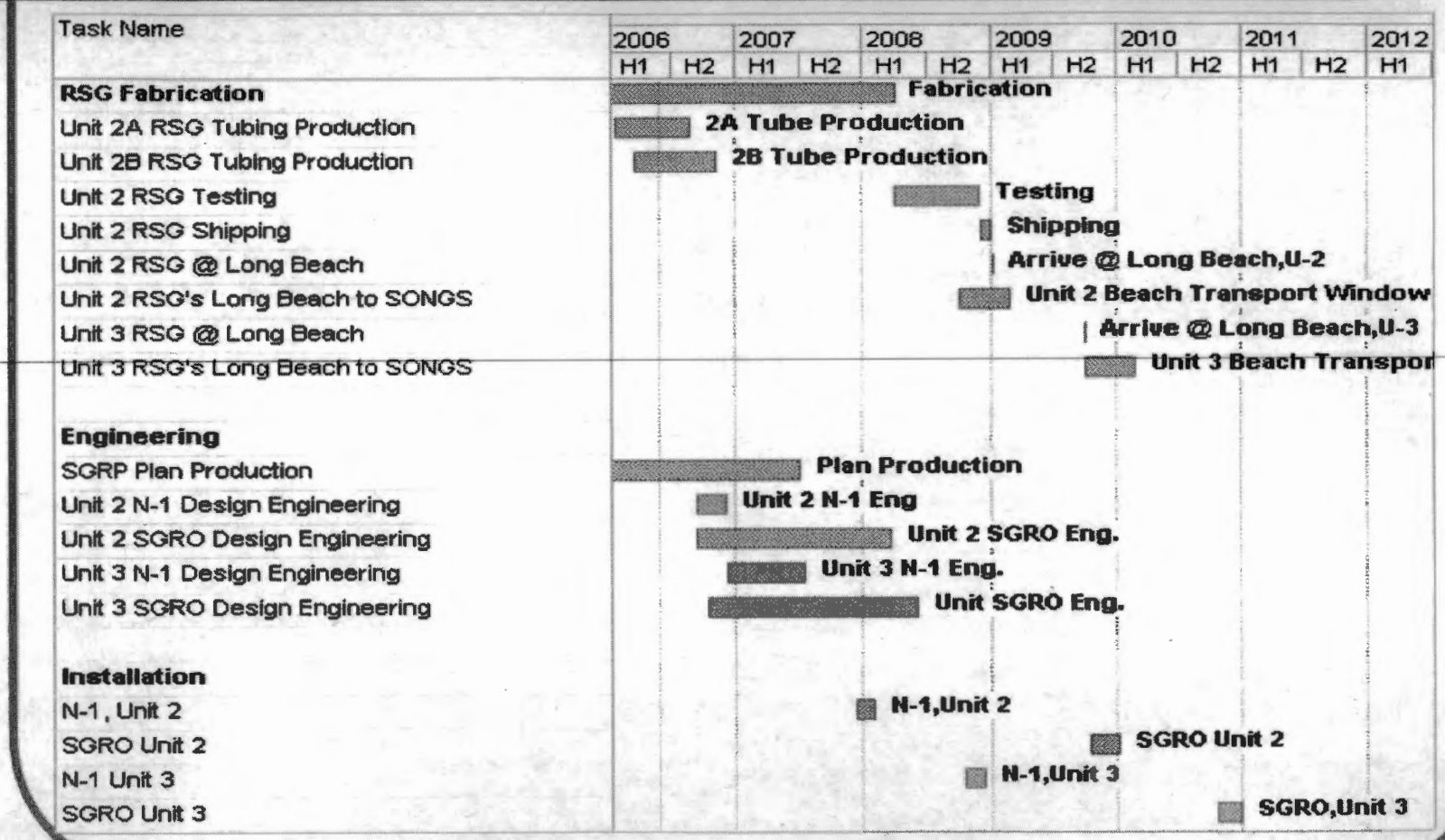


- Licensed to Operate: Until 2022
- Nuclear Steam System Supplier: Combustion Engineering
- Architect/Engineer: Bechtel
- Turbine Supplier: English Electric
- Unit Output: 1,150 Megawatts each
- ABB-CE Steam Generator, Model 3410, two S/G per unit
- I-600 MA Tubing

Unit 2 Tube Plugging Projections

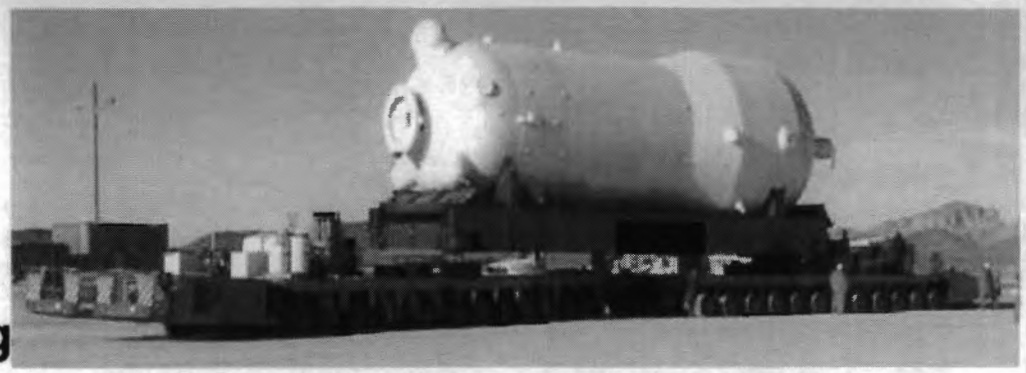
- Current Plugging Values
  - Unit 2: 13.5% Effective Plugging (Includes Sleeves)
  - Unit 3: 7.6% Plugging (No Sleeves)
- SG Inspections Before Cycle 16 Steam Generator Replacement Outage (SGRO)
  - Unit 2: Cycle 15
  - Unit 3: Cycle 14 & 15
- Plugging Limit is 21.4%
- Do Not Expect to Reach Plugging Limit

Schedule for SONGS Steam Generator Replacement



Benchmarking

- **Plant Benchmarking**
- **Fabricator Benchmarking**
- **Loan Employees**
- **Future Benchmarking Plans**



Palo Verde RSG Transport

➢ Recently Completed SGRO

- Palo Verde 1 & 2
- Beaver Valley 1
- ANO
- Callaway

➢ Future SGRO

- Ft. Calhoun
- Watts Bar
- Comanche Peak
- Palo Verde 3

SONGS TEAM

Current Status

- CPUC Application for Steam Generator Replacement Project (SGRP) Submitted February 2004
- Estimated Cost at $680m (2004 $)
- CPUC Decision December 2005
- EIX Board Accepted CPUC Decision March 2006

Licensing

- Will Be Implemented Under 10CFR 50.59
- No Power Uprate
- Associated Technical Specification Changes
  - Identification 2007

Replacement Steam Generators (RSG) Fabrication

Mitsubishi Heavy Industries Kobe, Japan

- Contract Award September 28, 2004

2A Primary

2A Secondary

RSG 2ATube Sheet

Mitsubishi Heavy Industries, Kobe, Japan
MHI
RSG Fabrication Facility
Engineering Offices

Oversight

- Design Reviews
- Technical Meetings (SONGS, Kobe)
- SCE Resident Personnel @ Kobe
- Special Engineering Visits
- Readiness Reviews
- Independent Inspections
- Audits

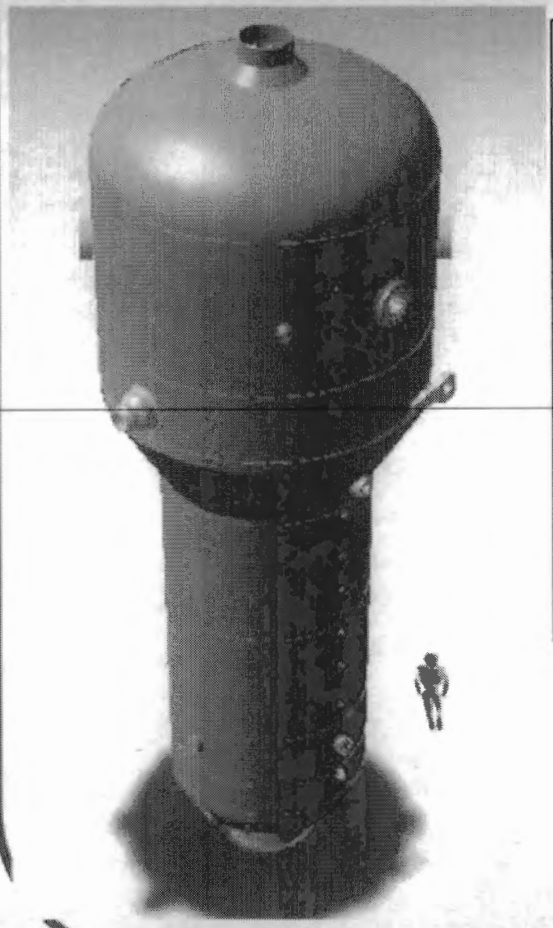
SONGS 2B Channel Head

Some Key Design Improvements

- Larger Surface Area
- Alloy 690 Thermally Treated Tubing
- Improved AVB Design
- Integral Steam Nozzle
- Improved Material for Tube Supports
- Forged Shell

S/G 3A Lower and Middle Shell
S/G 2A Balance Ring, Extension Ring, & Tubesheet

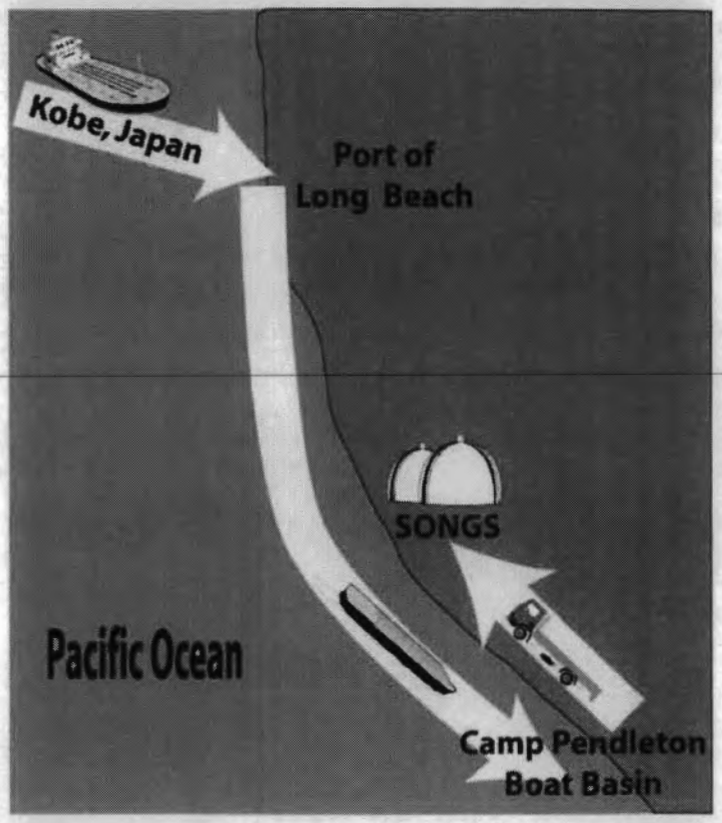
Replacement Steam Generators



	Original	RSG
Weight	620 tons	643.6 Tons
Height	65'6"	65'6"
Upper Section Diameter	22 feet	22 feet
Tubes	9,350 per SG	9,727 per SG
	¾ inch diameter	

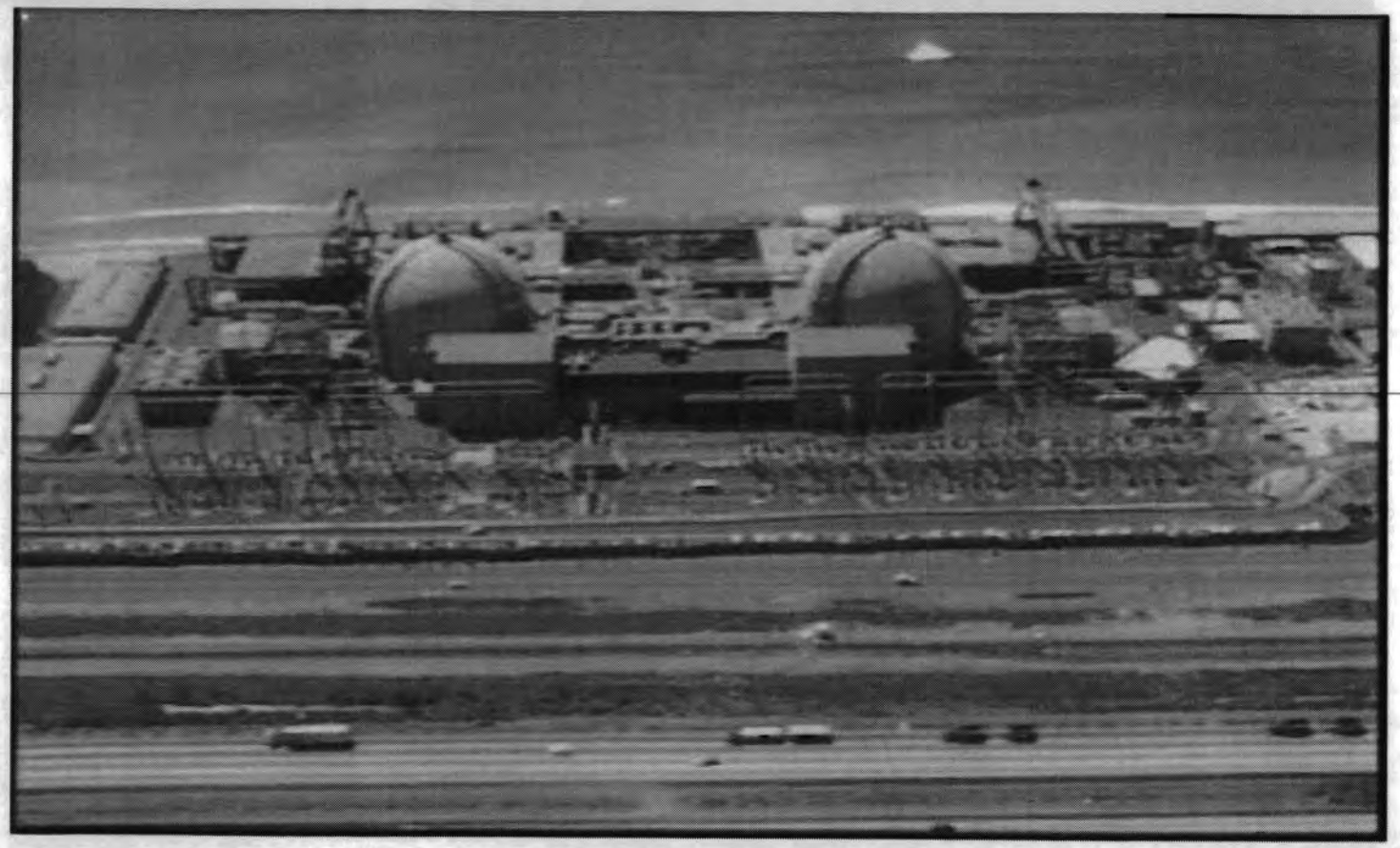
RSG Transportation to SONGS

- Heavy Lift Cargo Ship from Japan to Port of Long Beach
- Ocean Barge from Long Beach to Camp Pendleton
- Heavy Transport Vehicle from Camp Pendleton to SONGS

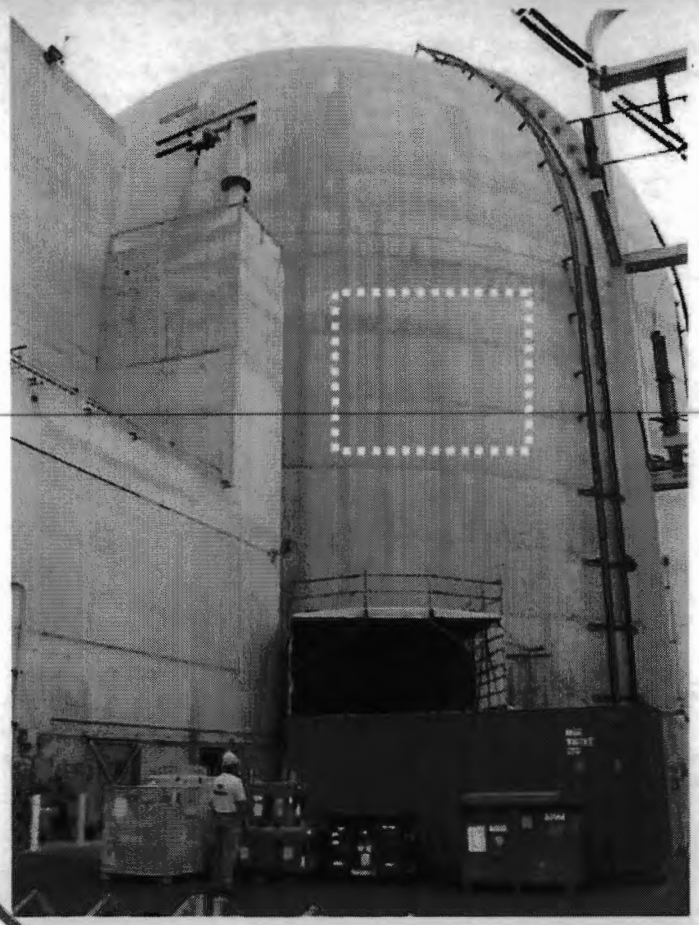


SONGS SGR TEAM

Key Implementation Considerations
Compact Site/Space Limitations

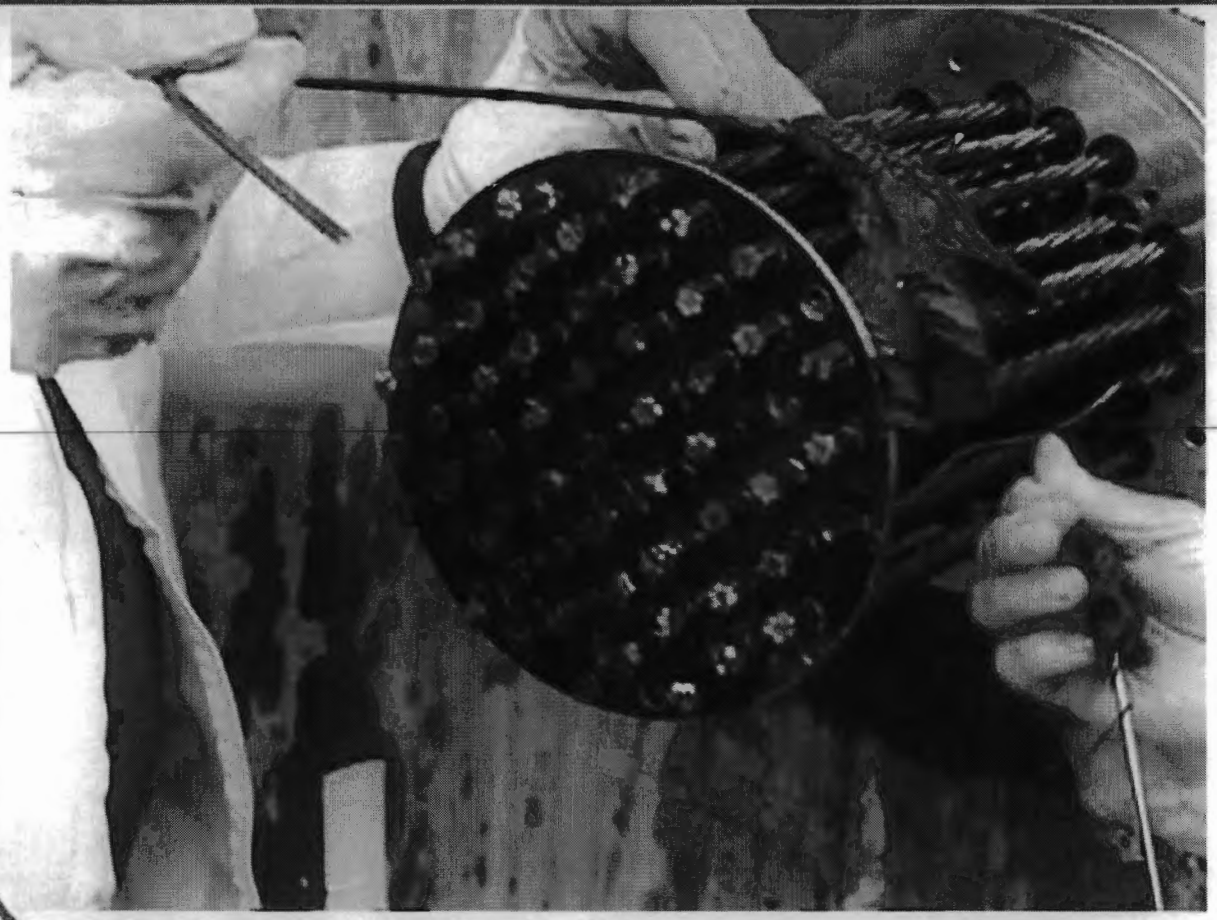


Key Implementation Considerations
Containment Penetration



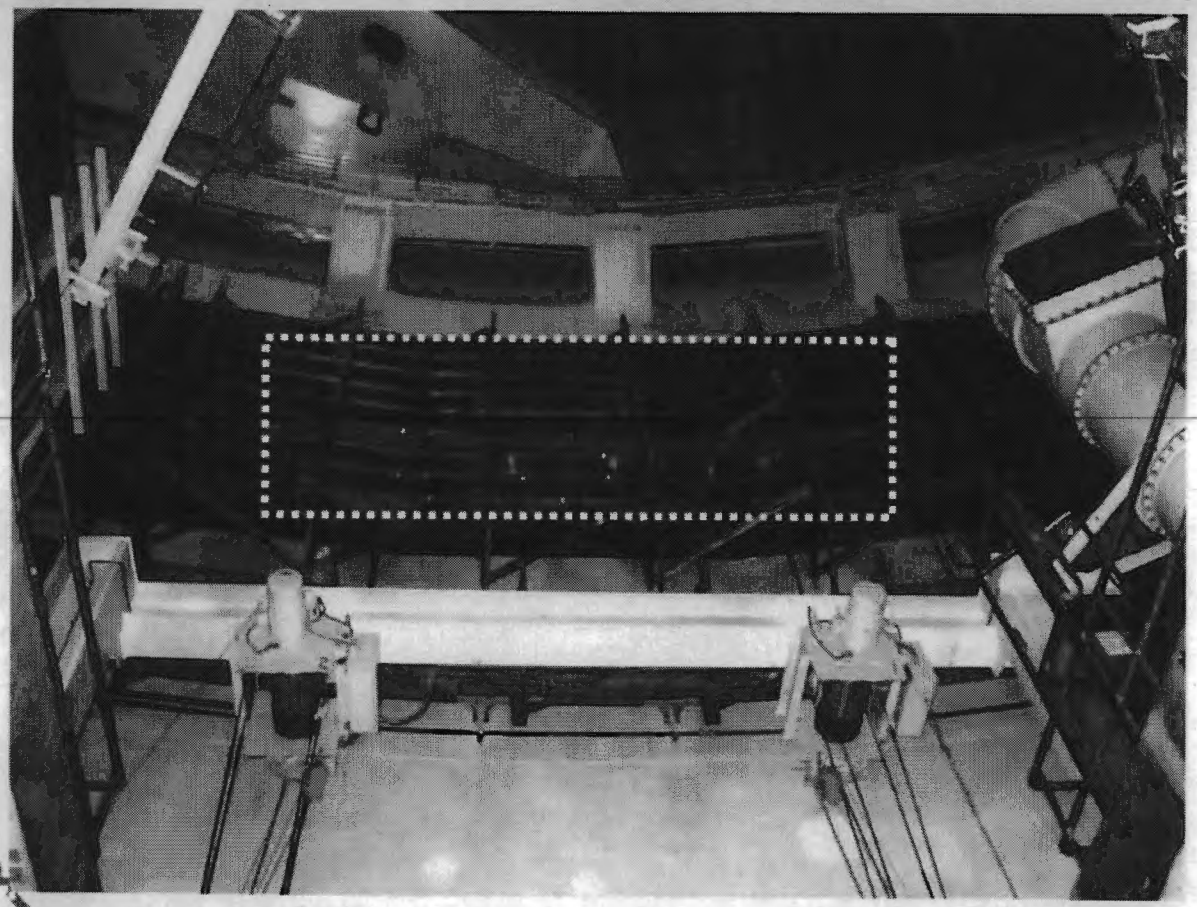
- 28′ x 28′ Opening
- 33.5′ Above Ground Level
- Over Equipment Hatch
- 4 ft Thick, Reinforced Wall
- 100 Cubic Yards Concrete
- Approximately 50 Tendons Will Be Removed

Key Implementation Considerations
Containment Tendon Design

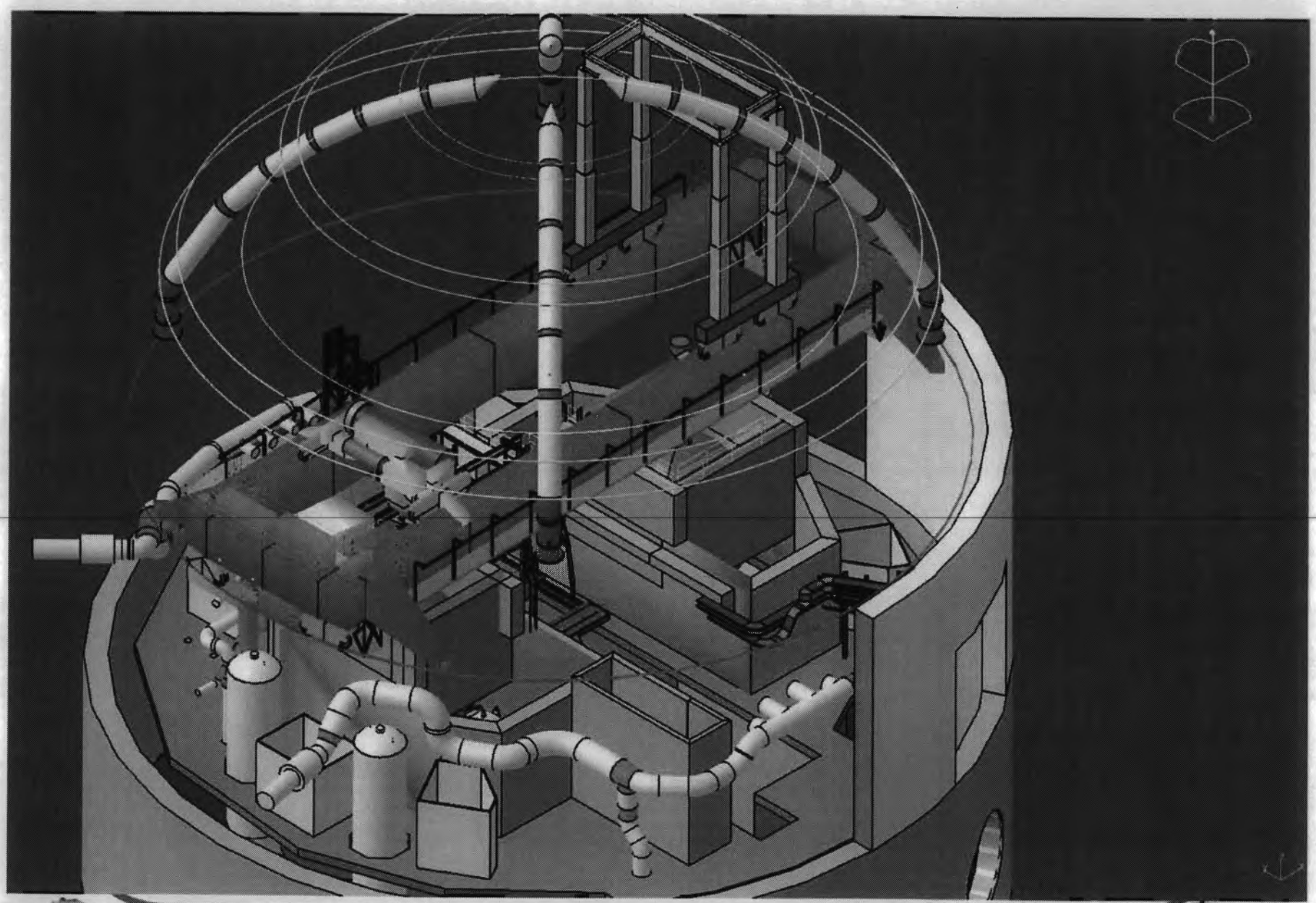


Rancho Seco Tendon Validation Test

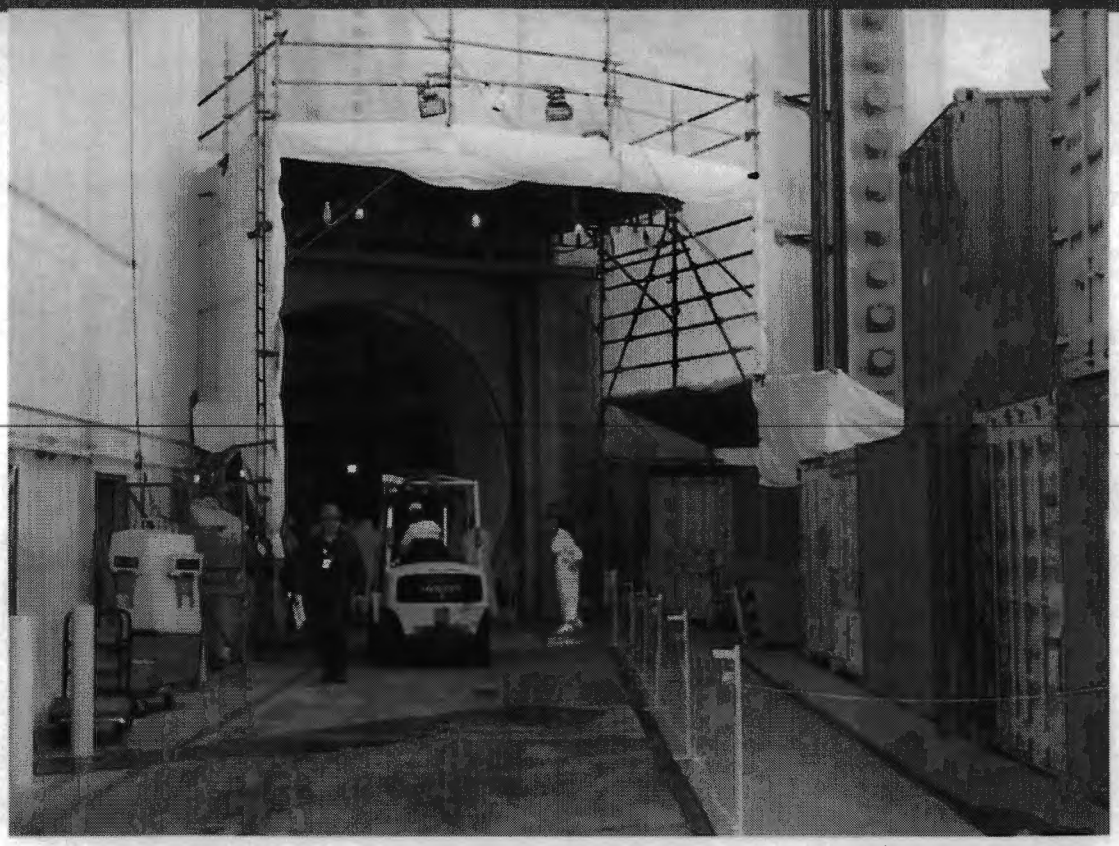
Key Implementation Considerations
Containment Interferences



Cable Trays Affected by Containment Breach



Key Implementation Considerations
Installation Contractor



Equipment hatch during a normal refueling outage

Bechtel
Awarded Installation Contract
December 2005

Original SONGS AE

Current Maintenance Contractor for SONGS

Significant SGR Experience

Early Project Involvement

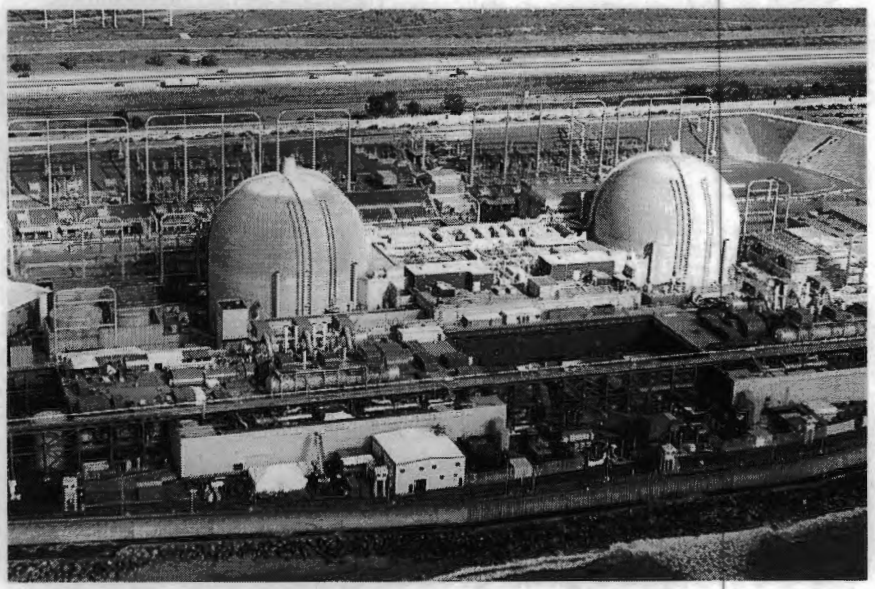
Original Steam Generator Disposal

- Disposal of OSG's Offsite Is Required Due to SONGS Compact Site
  - OSG's Large Size Requires Segmentation to Facilitate Shipping
  - Disposal at Energy Solutions, LLC, (formerly Envirocare of Utah, LLC) Planned

Q & A

Voice of OC | (http://voiceofoc.org/2012/05/nrc-releases-missing-record-on-san-onofre-plant-design-changes/)

NRC Releases Missing Record on San Onofre Plant Design Changes



San Onofre Nuclear Generating Station (Photo Credit: Enformable)

By NICK GERDA May 15, 2012 at 4:42 PM

Federal regulators Monday released a key presentation on generator design changes at San Onofre Nuclear Generating Station that had been missing from records (http://voiceofoc.org/healthy_communities/article_7ffff700-9bd7-11e1-a8ee-0019bb2963f4.html) of the public meeting where it was presented.

The 2006 presentation by plant operator Southern California Edison to the U.S. Nuclear Regulatory Commission was billed as an overview of its plans to modify and install replacement generators.

The presentation, however, didn't include major design alterations that an outside nuclear expert has concluded were responsible for a minor radiation leak in late January. And several of the changes that were not in the presentation should have triggered a much more thorough review by the government, according to the expert, Arnie Gundersen.

San Onofre remains shut down after the Jan. 31

RELATED LINKS

- Expert Says Edison Misled Regulators on San Onofre Changes (http://voiceofoc.org/healthy_communities/article_ca8c632a-85bc-11e1-a74c-001a4bcf887a.html)

radiation leak, which officials say was caused by extremely rapid wear (http://voiceofoc.org/oc_coast/san_clemente/article_abfd7168-8064-11e1-b0ad-001a4bcf887a.html) to generator tubes. Edison now believes the rapid wear, in turn, was likely caused by a "design defect (http://www.utsandiego.com/news/2012/may/03/san-onofre-reactors-could-return-in-june/?page=2#article)."

The presentation's release came the next business day after Voice of OC reported (http://voiceofoc.org/healthy_communities/article_7ffff700-9bd7-11e1-a8ee-0019bb2963f4.html) that a month after regulators told Gundersen that they couldn't find the document there still was no sign of it.

Voice of OC's calls and emails in recent days to both Edison and NRC officials on the issue have not been returned.

The plant's replacement steam generators, which were installed in 2010 and 2011, were supposed to last decades. But tubes vibrated and rubbed against each other, wearing them down much more quickly than expected, according to Edison and the NRC.

In just a year, tubes in Unit 3 had worn down so much that a break occurred, causing the small leak of radiation into the nonradioactive side of the plant. The leak posed no risk to workers or the public, officials say.

The same type of wear was found in the other unit's generators, and more than 1,300 tubes have been taken out of service since the leak investigation began.

Since March, Gundersen, who was hired by the environmental group Friends of the Earth to conduct an independent investigation, has charged that the wear was caused by significant design changes Edison made to the replacement generators, stemming from the addition of nearly 400 tubes in each of the four generators.

Gundersen also alleges the company mislead the government (http://voiceofoc.org/healthy_communities/article_ca8c632a-85bc-11e1-a74c-001a4bcf887a.html) about the changes in order to avoid a thorough

- NRC Shuts Down San Onofre Nuclear Plant Indefinitely (http://voiceofoc.org/oc_south/article_353ace5a-7938-11e1-9c98-0019bb2963f4.html)
- Regulator Vows Full Accounting of Tube Failure at San Onofre (http://voiceofoc.org/oc_coast/san_clemente/article_abfd7168-8064-11e1-b0ad-001a4bcf887a.html)
- Tubes Fail Pressure Test at San Onofre Plant (http://voiceofoc.org/countywide/this_just_in/article_c7cc6360-6ef1-11e1-bed3-0019bb2963f4.html)

independent analysis of their safety risks.

Edison, meanwhile, insists it was fully transparent with regulators about the alterations. The company and the NRC also say it's too early to draw conclusions about the ultimate cause of the wear.

In a third report (http://libcloud.s3.amazonaws.com/93/01/3/1442/SO_Steam_Generator_Analysis_May.pdf) issued Tuesday, Gundersen states the changes met nearly 40 triggers, each of which, on their own, requires a thorough review of safety risks by the government.

In its 2006 presentation to the NRC, however, Edison declares that the generator replacement "will be implemented under 10CFR 50.59," a process without government analysis of safety risks. The federal nuclear safety agency ultimately accepted that path, though it's unclear at this point whether regulators were provided with complete information.

Edison's presentation covers six design changes but neglects to mention key modifications that Gundersen says were responsible for the excessive tube wear.

One of the changes was the removal of the stay cylinder, which Gundersen describes as "the main structural pillar" in the generator. That alteration, combined with other changes, created the vibrations and rubbing that caused excessive wear, Gundersen wrote.

After saying in early April it would provide details on when it informed the NRC of the specific changes mentioned by Gundersen, Edison hasn't returned several follow-up messages on the topic.

Neither Edison nor the NRC had a response to Gundersen's latest report by mid-morning on Tuesday.

Gundersen noticed the missing presentation and asked the NRC to provide it on April 12, emails show. The next day, the agency replied that it couldn't find it and would seek a copy from Edison.

Contacted on Thursday for information about why the record was missing, representatives for Edison and the NRC told Voice of OC they needed time to research the issue. They had nothing further to add on Friday, and did not return messages Monday afternoon seeking more information.

However, the LA Times reported (http://latimesblogs.latimes.com/lanow/2012/05/edison-san-onofre-design-changes.html) Monday evening that the presentation was released earlier in the day.

The NRC recently said it is reviewing the generator approval process to "ensure that [Edison] followed proper procedures."

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