Richard J. St. Onge
Director, Nuclear Regulatory Affairs and
Emergency Planning

Proprietary Information
Withhold from Public Disclosure

February 14, 2013

10 CFR 50.4

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

Subject: **Docket No. 50-361**
Supplemental Document Submittal
Regarding Confirmatory Action Letter Response
(TAC No. ME 9727)
San Onofre Nuclear Generating Station, Unit 2

References:

1. Letter from Mr. Elmo E. Collins (USNRC) to Mr. Peter T. Dietrich (SCE), dated March 27, 2012, Confirmatory Action Letter 4-12-001, San Onofre Nuclear Generating Station, Units 2 and 3, Commitments to Address Steam Generator Tube Degradation

2. Letter from Mr. Peter T. Dietrich (SCE) to Mr. Elmo E. Collins (USNRC), dated October 3, 2012, Confirmatory Action Letter – Actions to Address Steam Generator Tube Degradation, San Onofre Nuclear Generating Station, Unit 2

Dear Sir or Madam,

On March 27, 2012, the Nuclear Regulatory Commission (NRC) issued a Confirmatory Action Letter (CAL) (Reference 1) to Southern California Edison (SCE) describing actions that the NRC and SCE agreed would be completed to address issues identified in the steam generator tubes of San Onofre Nuclear Generating Station (SONGS) Units 2 and 3. In a letter to the NRC dated October 3, 2012 (Reference 2), SCE reported completion of the Unit 2 CAL actions and included a Return to Service Report that provided details of their completion.

The purpose of this letter is to respond to an information request from the NRC associated with the Unit 2 CAL response. Specifically, the NRC requested submittal of the 10 CFR 50.59 screenings and evaluations related to the SONGS replacement steam generators (RSGs). Enclosures 1-2 of this submittal provide the current revision of the proprietary version of the Unit 2 and Unit 3 screenings as listed in the enclosure table below. No proprietary information was identified in the Unit 2 and 3 evaluations. Because the enclosures contain information that is proprietary, SCE requests this information be withheld from public disclosure in accordance with 10 CFR 2.390(a)(4). Enclosures 3 and 4 provide notarized affidavits from Mitsubishi Heavy Industries (MHI) and Westinghouse Electric Company (WEC) which set forth the basis on which the information in Enclosures 1-2 may be withheld from public disclosure and addresses with specificity the considerations listed by paragraph (b)(4) of 10 CFR 2.390. Enclosures 5-8 of this submittal provide the current revision of the non-proprietary version of the Unit 2 and Unit 3 10 CFR 50.59 screenings and evaluations.

P.O. Box 128
San Clemente, CA 92672

IE36
NRR

Enclosure	Document Title	Tracking Number
1	Unit 2 10 CFR 50.59 Screening (Proprietary, P)	800175663-0530
2	Unit 3 10 CFR 50.59 Screening (P)	800175664-0190
3	Affidavit from MHI for the RSG 10 CFR 50.59 Screenings and Evaluations (Enclosures 1-2)	
4	Affidavit from WEC for the RSG 10 CFR 50.59 Screenings and Evaluations (Enclosures 1-2)	
5	Unit 2 10 CFR 50.59 Screening (Non-Proprietary, NP)	800175663-0530
6	Unit 2 10 CFR 50.59 Evaluation (NP)	800175663-0520
7	Unit 3 10 CFR 50.59 Screening (NP)	800175664-0190
8	Unit 3 10 CFR 50.59 Evaluation (NP)	800175664-0170

There are no new regulatory commitments contained in this letter. If you have any questions or require additional information, please call me at (949) 368-6240.

Sincerely,

Enclosures: As described in the table above

cc: E. E. Collins, Regional Administrator, NRC Region IV
J. R. Hall, NRC Project Manager, SONGS Units 2 and 3
G. G. Warnick, NRC Senior Resident Inspector, SONGS Units 2 and 3
R. E. Lantz, Branch Chief, SONGS Project Branch, NRC Region IV
G. E. Werner, Inspection & Assessment Lead, SONGS Project Branch

ENCLOSURE 3

Affidavit from MHI for the RSG
10 CFR 50.59 Screenings and Evaluations

(Non-Proprietary)

MITSUBISHI HEAVY INDUSTRIES, LTD.

AFFIDAVIT

I, Jinichi Miyaguchi, state as follows:

1. I am Director, Nuclear Plant Component Designing Department, of Mitsubishi Heavy Industries, Ltd. ("MHI"), and have been delegated the function of reviewing the referenced documentations to determine whether they contain MHI's information that should be withheld from public disclosure pursuant to 10 C.F.R. § 2.390 (a)(4) as trade secrets and commercial or financial information that is privileged or confidential.

2. In accordance with my responsibilities, I have reviewed the following documentations and have determined that they contain MHI proprietary information that should be withheld from public disclosure. Those pages containing proprietary information have been bracketed with an open and closed bracket as shown here "[ ]" / and should be withheld from public disclosure pursuant to 10 C.F.R. § 2.390 (a)(4).

SCE's documents

- Root Cause Evaluation 201836127
  Unit 3 Steam Generator Tube Leak and Tube-to-Tube Wear Condition Report
  San Onofre Nuclear Generating Station (SONGS)
- Root Cause Evaluation NN 201843216
  Steam Generator Tube Wear
  San Onofre Nuclear Generating Station, Unit 2
- 10CFR50.59 Evaluation, Screening
  NECP 800175663
  Steam Generator Replacement Mstr ECP U2
- 10CFR50.59 Evaluation, Screening
  NECP 800175664
  Steam Generator Replacement Mstr ECP U3

AREVA's documents

- SONGS Unit 2 Probability of FEI Operational Assessment RAI Responses
  51-9197672

MHI's document

- L5-04GA564
  Tube wear of Unit 3 RSG – Technical Evaluation Report

3. The information identified as proprietary in the documents have in the past been, and will continue to be, held in confidence by MHI and its disclosure outside the company is limited to regulatory bodies, customers and potential customers, and their agents, suppliers, and licensees, and others with a legitimate need for the information, and is always subject to suitable measures to protect it from unauthorized use or disclosure.

4. The basis for holding the referenced information confidential is that they describe unique design, manufacturing, experimental and investigative information developed by MHI and not used in the exact form by any of MHI's competitors. This information was developed at significant cost to MHI, since it is the result of an intensive MHI effort.

5. The referenced information was furnished to the Nuclear Regulatory Commission ("NRC") in confidence and solely for the purpose of information to the NRC staff.

6. The referenced information is not available in public sources and could not be gathered readily from other publicly available information. Other than through the provisions in paragraph 3 above, MHI knows of no way the information could be lawfully acquired by organizations or individuals outside of MHI.

7. Public disclosure of the referenced information would assist competitors of MHI in their design and manufacture of nuclear plant components without incurring the costs or risks associated with the design and the manufacture of the subject component. Therefore, disclosure of the information contained in the referenced documents would have the following negative impacts on the competitive position of MHI in the U.S. and world nuclear markets:

   A. Loss of competitive advantage due to the costs associated with development of technologies relating to the component design, manufacture and examination. Providing public access to such information permits competitors to duplicate or mimic the methodology without incurring the associated costs.

   B. Loss of competitive advantage of MHI's ability to supply replacement or new heavy components such as steam generators.

I declare under penalty of perjury that the foregoing affidavit and the matters stated therein are true and correct to the best of my knowledge, information and belief.

Executed on this 1 day of February, 2013.

Jinichi Miyaguchi

Jinichi Miyaguchi,
Director- Nuclear Plant Component Designing Department
Mitsubishi Heavy Industries, LTD

Sworn to and subscribed

Before me this 1 day

of February, 2013

Masahiko Kubota

Notary Public

NOTARY
Arashimachi, Chuo-
Kobe, Jap

16 FEB. - 1. 2013

My Commission Expires ____________

登簿平成２５年第16号

認　　証

嘱託人　三菱重工業株式会社　原子力事業本部
原子力製造総括部　原子力機器設計部　部長　宮
口仁一　は本職の面前で添付書面に　署名　した。

Legal Affairs
NOTARY
44 Akashi

よって認証する。

平成２５年２月１日

本職役場に於て

神戸市中央区明石町４４番地

神戸地方法務局所属

公証人　窪田正彦

公証人役場

Registered Number 16 Date FEB. - 1, 2013

NOTARIAL CERTIFICATE

This is to certify that JINICHI MIYAGUCHI , Director-Nuclear Plant Component Designing Department MITSUBISHI HEAVY INDUSTRIES, LTD has affixed his signature in my very presence to the attached document.

Masahiko Kubota
Kobe District Legal Affairs Bureau
NOTARY
44 Akashimachi, Chuo-ku,
Kobe, Japan

Masahiko Kubota

MASAHIKO KUBOTA

Notary

44 Akashimachi, Chuo-Ku,

Kobe, Japan

Kobe District Legal Affairs Bureau

(面前法2)

ENCLOSURE 4

Affidavit from WEC for the RSG
10 CFR 50.59 Screenings and Evaluations

(Non-Proprietary)

Westinghouse

Westinghouse Electric Company
Nuclear Services
1000 Westinghouse Drive
Cranberry Township, Pennsylvania 16066
USA

U.S. Nuclear Regulatory Commission
Document Control Desk
11555 Rockville Pike
Rockville, MD 20852

Direct tel: (412) 374-4643
Direct fax: (724) 720-0754
e-mail: greshaja@westinghouse.com
Proj letter: CONO-13-15

CAW-13-3619

February 7, 2013

APPLICATION FOR WITHHOLDING PROPRIETARY
INFORMATION FROM PUBLIC DISCLOSURE

Subject: Markup of Unit 2 and 3 SONGS Replacement Steam Generator 50.59 Screens and Evaluations (Proprietary)

The proprietary information for which withholding is being requested in the above-referenced report is further identified in Affidavit CAW-13-3619 signed by the owner of the proprietary information, Westinghouse Electric Company LLC. The affidavit, which accompanies this letter, sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of 10 CFR Section 2.390 of the Commission's regulations.

Accordingly, this letter authorizes the utilization of the accompanying affidavit by San Onofre Nuclear Generating Station.

Correspondence with respect to the proprietary aspects of the application for withholding or the Westinghouse affidavit should reference CAW-13-3619, and should be addressed to James A. Gresham, Manager, Regulatory Compliance, Westinghouse Electric Company, Suite 428, 1000 Westinghouse Drive, Cranberry Township, Pennsylvania 16066.

Very truly yours,

J. A. Gresham

James A. Gresham, Manager
Regulatory Compliance

Enclosures

CAW-13-3619

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF BUTLER:

Before me, the undersigned authority, personally appeared James A. Gresham, who, being by me duly sworn according to law, deposes and says that he is authorized to execute this Affidavit on behalf of Westinghouse Electric Company LLC (Westinghouse), and that the averments of fact set forth in this Affidavit are true and correct to the best of his knowledge, information, and belief:

James A. Gresham, Manager
Regulatory Compliance

Sworn to and subscribed before me
this 7th day of February 2013

Notary Public

COMMONWEALTH OF PENNSYLVANIA
Notarial Seal
Anne M. Stegman, Notary Public
Unity Twp., Westmoreland County
My Commission Expires Aug. 7, 2016
MEMBER, PENNSYLVANIA ASSOCIATION OF NOTARIES

(1) I am Manager, Regulatory Compliance, in Nuclear Services, Westinghouse Electric Company LLC (Westinghouse), and as such, I have been specifically delegated the function of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear power plant licensing and rule making proceedings, and am authorized to apply for its withholding on behalf of Westinghouse.

(2) I am making this Affidavit in conformance with the provisions of 10 CFR Section 2.390 of the Commission's regulations and in conjunction with the Westinghouse Application for Withholding Proprietary Information from Public Disclosure accompanying this Affidavit.

(3) I have personal knowledge of the criteria and procedures utilized by Westinghouse in designating information as a trade secret, privileged or as confidential commercial or financial information.

(4) Pursuant to the provisions of paragraph (b)(4) of Section 2.390 of the Commission's regulations, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld.

(i) The information sought to be withheld from public disclosure is owned and has been held in confidence by Westinghouse.

(ii) The information is of a type customarily held in confidence by Westinghouse and not customarily disclosed to the public. Westinghouse has a rational basis for determining the types of information customarily held in confidence by it and, in that connection, utilizes a system to determine when and whether to hold certain types of information in confidence. The application of that system and the substance of that system constitutes Westinghouse policy and provides the rational basis required.

Under that system, information is held in confidence if it falls in one or more of several types, the release of which might result in the loss of an existing or potential competitive advantage, as follows:

(a) The information reveals the distinguishing aspects of a process (or component, structure, tool, method, etc.) where prevention of its use by any of

Westinghouse's competitors without license from Westinghouse constitutes a competitive economic advantage over other companies.

(b) It consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), the application of which data secures a competitive economic advantage, e.g., by optimization or improved marketability.

(c) Its use by a competitor would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing a similar product.

(d) It reveals cost or price information, production capacities, budget levels, or commercial strategies of Westinghouse, its customers or suppliers.

(e) It reveals aspects of past, present, or future Westinghouse or customer funded development plans and programs of potential commercial value to Westinghouse.

(f) It contains patentable ideas, for which patent protection may be desirable.

There are sound policy reasons behind the Westinghouse system which include the following:

(a) The use of such information by Westinghouse gives Westinghouse a competitive advantage over its competitors. It is, therefore, withheld from disclosure to protect the Westinghouse competitive position.

(b) It is information that is marketable in many ways. The extent to which such information is available to competitors diminishes the Westinghouse ability to sell products and services involving the use of the information.

(c) Use by our competitor would put Westinghouse at a competitive disadvantage by reducing his expenditure of resources at our expense.

(d) Each component of proprietary information pertinent to a particular competitive advantage is potentially as valuable as the total competitive advantage. If competitors acquire components of proprietary information, any one component may be the key to the entire puzzle, thereby depriving Westinghouse of a competitive advantage.

(e) Unrestricted disclosure would jeopardize the position of prominence of Westinghouse in the world market, and thereby give a market advantage to the competition of those countries.

(f) The Westinghouse capacity to invest corporate assets in research and development depends upon the success in obtaining and maintaining a competitive advantage.

(iii) The information is being transmitted to the Commission in confidence and, under the provisions of 10 CFR Section 2.390, it is to be received in confidence by the Commission.

(iv) The information sought to be protected is not available in public sources or available information has not been previously employed in the same original manner or method to the best of our knowledge and belief.

(v) The proprietary information sought to be withheld in this submittal is that which is contained in attached markups "Markup of Unit 2 and 3 SONGS Replacement Steam Generator 50.59 Screens and Evaluations" (Proprietary), for submittal to the Commission, being transmitted by San Onofre Nuclear Generating Station letter and Application for Withholding Proprietary Information from Public Disclosure, to the Document Control Desk.

This information is part of that which will enable Westinghouse to:

(a) Provide replacement steam generators.

Further this information has substantial commercial value as follows:

(a) Westinghouse plans to sell the use of similar information to its customers for the purpose of providing replacement steam generators.

(b) Westinghouse can sell support and defense of replacement steam generators.

(c) The information requested to be withheld reveals the distinguishing aspects of a design which was developed by Westinghouse.

Public disclosure of this proprietary information is likely to cause substantial harm to the competitive position of Westinghouse because it would enhance the ability of competitors to provide similar technical evaluation justifications and licensing defense services for commercial power reactors without commensurate expenses. Also, public disclosure of the information would enable others to use the information to meet NRC requirements for licensing documentation without purchasing the right to use the information.

The development of the technology described in part by the information is the result of applying the results of many years of experience in an intensive Westinghouse effort and the expenditure of a considerable sum of money.

In order for competitors of Westinghouse to duplicate this information, similar technical programs would have to be performed and a significant manpower effort, having the requisite talent and experience, would have to be expended.

Further the deponent sayeth not.

PROPRIETARY INFORMATION NOTICE

Transmitted herewith are proprietary and/or non-proprietary versions of documents furnished to the NRC in connection with requests for generic and/or plant-specific review and approval.

In order to conform to the requirements of 10 CFR 2.390 of the Commission's regulations concerning the protection of proprietary information so submitted to the NRC, the information which is proprietary in the proprietary versions is contained within brackets, and where the proprietary information has been deleted in the non-proprietary versions, only the brackets remain (the information that was contained within the brackets in the proprietary versions having been deleted). The justification for claiming the information so designated as proprietary is indicated in both versions by means of lower case letters (a) through (f) located as a superscript immediately following the brackets enclosing each item of information being identified as proprietary or in the margin opposite such information. These lower case letters refer to the types of information Westinghouse customarily holds in confidence identified in Sections (4)(ii)(a) through (4)(ii)(f) of the affidavit accompanying this transmittal pursuant to 10 CFR 2.390(b)(1).

ENCLOSURE 5

Unit 2 – 10 CFR 50.59 Screening
Completed On: 09/18/2009

(Non-Proprietary)

EDISON

SONGS

Operations Detail

Order: NECP 800175663
Description: Steam Generator Replacement Mstr ECP U2

Operation No: 0530 **Sub Op:** **Op Wrkctr:** ED_SGD SGRP Design
Description: 10CFR50.59 Screening
Work: 4.0 H **Number:** 1 **Duration:** 4.0 H
Risk level: **Potential Rx Impact:**
Risk Type:
RiskComments:

Operation Details:

10CFR50.59 Screening
(Previous Operation 800175663-0510)

NOTE: Completing a 50.59 Screen requires PQS T3EN98

SECTION I: GENERAL INFORMATION (See Keypoints Section I)

1. Primary Document Number and Title:

NECP 800071702, Steam Generator Replacement Unit 2, ASC D0018051

2. Description of Proposed Activity:

The proposed activity replaces the design disclosure documentation and reference documentation for the SONGS Units 2 Original Steam Generators (OSGs) with that for the Replacement Steam Generators (RSGs) and performs functional testing of the RSGs. This replacement and testing is required as a result of physically replacing the OSGs with the RSGs Having the OSGs replaced with the RSGs will improve efficiency and reliability of Unit 2 by replacing a large number of plugged or otherwise degraded heat transfer tubes in each OSG with new tubes made from thermally-treated Alloy 690, which is less susceptible to degradation than the mill-annealed Alloy 600 material used for OSG heat transfer tubing. Replacement of the steam generators is a replacement in-kind in terms of an overall fit, form and function with no, or minimal, permanent modifications to the plant systems, structures or components (SSCs).

The proposed activity covered by this screen includes the following:

Replacing the Original Steam Generator (OSG) documentation with the Replacement Steam Generator (RSG) documentation, which includes voiding/superseding the original Combustion Engineering/Westinghouse Electric Co. (CE/WEC) documents, as appropriate, and adding the new Mitsubishi Heavy Industries (MHI) design documents to the database.
·Revising affected design bases documents (DBDs, UFSAR, calculations, etc.).
·Revising other affected documents (90000 series documents, EQDPs, System Descriptions, Licensee Controlled Specifications (LCS), etc.).
·Functional (performance) testing of the RSGs after installation.

Operation Details Cont'd:

NOTE:
Activities associated with the physical removal of the OSGs and installation of the RSGs, such as creation and restoration of a containment building temporary construction opening, replacement of the steam generator insulation, re-calibration of the steam generator level transmitters or Cycle 16 Reload Analysis, are not addressed herein, but instead are addressed in the specific NECPs governing these activities.

The RSG design bases are consistent with those of the OSGs, except for the specific values of the major parameters, and are as follows (see UFSAR Sections 4.4, 5.1, 5.2, 5.4 and 10.1):

The RSGs are designed and fabricated as ASME B&PV Code Section III, Class 1 components, based on the 1998 Edition through 2000 Addenda. The OSGs are designed and fabricated as Class A components in conformance with the ASME B&PV Code, Section III, 1971 Edition plus Addenda through summer 1971.
Each RSG is designed to remove the rated heat load of 1729 MWt from the reactor coolant system (RCS), which is the same as the OSG current operating heat load.
·Each RSG is designed to remove the rated heat load with up to 8% (779) tubes plugged, which is the same as the design plugging limit for the OSGs.
Each RSG is designed to produce 7.588E6 lb/hr (vs. 7.414E6 lb/hr for OSGs) of 833 psia (vs. 900 psia for OSGs) saturated steam with less than 0.1% (vs. 0.2% for OSGs) moisture content when supplied with feedwater at 442oF.

NOTES:
The saturated steam pressure of 833 psia is the best estimate prtessure at the steam generator outlet nozzle with the reactor coolant inlet (Thot) temperature of 598 oF (corresponding to the Tcold of 541.3oF), reactor coolant best estimate flow rate (79.79E6 lb/hr), no (0%) tubes plugged and an assumed fouling factor ([ ] ft2hroF/Btu). The actual pressure will vary depending on the actual values of these parameters during plant operation.
The RSGs are qualified to operate in the Thot range from 598 to 611oF, which corresponds to the Tcold range from 541.3 to 555.4oF. However, NECP 800071702 and this 50.59 screen only allows the plant to operate up to Thot <= 598oF, as additional analyses are required to be performed to evaluate the rest of the RCS and support systems' ability to operate above Thot > 598oF.

The major physical differences between the RSGs and OSGs are as follows:

The RSGs have a greater number of tubes (9,727 vs. 9,350) and a larger heat transfer surface area than the OSGs ([ ] ft2 vs. ~ [ ] ft2).
The RSG reactor coolant volume is greater than the OSG volume (2003 ft3 vs. 1895 ft3).
·The RSG tube wall thickness is less than the wall thickness of the OSG tubes (0.0429 in. vs. 0.048 in.).
The RSG tubes are Alloy 690 (thermally-treated) while the OSG tubes are Alloy 600 (mill-annealed).
The RSG feedwater ring is fabricated from erosion-corrosion resistant Cr-Mo alloy steel with Alloy 690 TT fittings, whereas the OSG feedwater ring is made of carbon steel (with the exception of the flow distribution box).

All RSG tubes are U-bend shape, whereas the OSG tubes have both U-bend shape (inner rows of the tube bundle) and square-bend shape (outer rows of the tube bundle).

Operation Details Cont'd:

·The RSG channel head has a flat bottom, thicker divider plate, as compared to the OSGs, and no stay cylinder.
The RSGs have integral grooves on all primary nozzles for nozzle dam mounting, whereas the OSGs do not.
The RSG tube-to-tubesheet joints are seal-welded and hydraulically expanded with a one-step mechanical roll at the primary face, while the OSG joints are seal-welded and explosively expanded.
The RSG tube supports consist of 7 broached tube support plates in the straight-leg region and anti-vibration bars in the U-bend region, while the OSG tube supports consist of the egg-crate type supports in the straight-leg region and batwings and vertical strips in the U-bend region.
The RSG feedwater ring has a "gooseneck" preventing water hammer and employs perforated spray nozzles mounted on top of the ring, whereas the OSGs do not have the "gooseneck and employ j-nozzles.
The RSGs have [ ] moisture separators and [] banks of steam dryers each, whereas the OSGs have [ ] moisture separators and [ ] steam dryers.
The RSG steam nozzle is an integrally forged part of the upper head (i.e. there is no weld seam between the nozzle and the head) and has 7 integral venturi nozzles designed to limit the steam flow and reduce the rate of energy release into the containment during a postulated main steam line break (MSLB) inside containment. The OSGs have no such flow limiting device.
The number and orientation of the RSG major nozzles and instrument taps are the same as for the OSGs, except that: the RSGs have one wide range level lower tap on the extension ring, whereas the OSGs have such a tap installed on the handhole cover; the circumferential orientation of two narrow range level taps on the RSGs is different from that on the OSGs to avoid interference between the taps and one of the lifting trunnions; the RSGs have one each secondary side sampling tap, dry lay-up nozzle, wet lay-up nozzle, and recirculation nozzle on the upper shell, whereas the OSGs have only the sampling tap; the RSGs have one blowdown/secondary drain nozzle located on the tubesheet, whereas the OSGs have such a nozzle on the lower shell above the tubesheet.
The RSGs have a greater number of secondary side handholes and inspection ports ([] at [] in. versus 2 at 6 in. and [ ] at [ ] in. versus none, respectively).
The RSGs have two lifting trunnions on their upper shell, separate from the secondary manways, for installation rigging. The OSGs had no separate lifting provision and utilized the secondary manwavs for installation rigging.
The RSG lower shell outside diameter is 174.6 in., whereas the OSG diameter is [ ] in.
·The RSGs have a larger primary manway inside diameter than the OSGs ([] in. vs. 16 in.).
RSG drv. flooded and operating weights are greater than the corresponding weights of the OSGs ([ ] lb. vs. [ ] lb., [ ] lb. vs. [ ] lb. and 1,548,700 lb. vs. [ ] lb, respectively).
·To maintain the SG within its fracture toughness limits, when the RSG secondary pressure is greater than 200 psig, the RSG steam generator secondary coolant (main or auxiliary feedwater) temperature must be less than 70oF, which is unchanged from the limit for the Unit 2 OSGs (70oF). [The Unit 2 OSG limitation of 70oF is based on a Nil Ductility Reference Temperature (RTndt) of [ ] oF, whereas the RSG limitation of 70oF is based on an RTndt of [ ] oF (ref. Unit 2 LCS 3.7.109).]
·A change was also made to LCS 3.7.109 to recognize that periodic surveillance of compliance to the secondary side temperature limit is not required during the time that the OSGs are drained for removal from the plant (with the core off-loaded) and the time the RSGs have been installed and may be pressurized (prior to reloading fuel in the core).

The proposed activity also involves updates of certain UFSAR-described analyses that demonstrate that with the RSGs installed the consequences of accidents would not result in exceeding the dose limits, or demonstrate that the intended design functions will be accomplished. In addition to the discussion of the RSG component, Sections I.3, II.1, II.3 and II.5 of this 50.59 Screen include the following subsections to address the effects of the proposed activity on the design functions, performance aspects, methods of evaluation, and technical

Operation Details Cont'd:

specifications/LCS of/for the plant systems, structures or components affected by the proposed activity:

Subsection (a) RCS Structural Integrity
Subsection (b) ECCS Performance
Subsection (c) Non-LOCA Transients
Subsection (d) Containment P-T Analysis
Subsection (e) Low Temperature Overpressure Protection
Subsection (f) RPS, ESFAS, COLSS and CPC
Subsection (g) NSSS Performance
Subsection (h) Non-Safety Related Control Systems Performance

3. Design Function(s) and/or Method(s) of Evaluation:

The design functions of the steam generators are to:

1. Function as a part of the reactor coolant pressure boundary (RCPB).
2. Transfer heat between the RCS and main steam system.
3. Remove heat from the RCS to achieve and maintain safe shutdown following design basis accidents (except for a large break LOCA) and other UFSAR-described events.

The design functions of the steam generator tubes and tube supports are to:

1. Limit tube flow-induced vibration and reactor coolant pump-induced vibration to acceptable levels during normal operating conditions.
2. Withstand blowdown forces from severance of a steam nozzle and ensure that ASME Code allowable stress limits are met.
3. Maintain acceptable ASME Code stress levels under design basis accident conditions (i.e., to prevent a tube rupture concurrent with other accidents, and to prevent multiple tube ruptures during a postulated single steam generator tube rupture event), and
4. Fnction as a part of the RCPB.

The UFSAR-described methods of evaluation for the steam generators are:

1. UFSAR Sections 3.9.1.2.2.1.11 and 3.9.1.2.2.2.3 state that the stress analysis and dynamic analysis of RCS components are performed by finite element method using the ANSYS computer program. Specifically, the finite element capabilities of the program are used to determine the primary plus secondary stresses and peak stresses due to normal and upset conditions and to evaluate the resulting stress intensities.
2. UFSAR 5.4.2.3.1.3 describes specific methods of evaluation for the Regulatory Guide (RG) 1.121 tube-wall thinning evaluation, using the following computer programs:
CEFLASH for blowdown analysis
STRUDL for tube displacement response analysis, and
ANSYS for tube stress analysis.

Subsection (a) RCS Structural Integrity

Operation Details Cont'd:

RCS design functions considered in this subsection are:
(1)Maintain structural integrity of the main coolant loop (MCL) piping (including tributary line nozzles and partial penetration nozzles), surge line (including partial penetration nozzles), reactor vessel (RV) and its supports, reactor coolant pumps (RCPs) and their supports, steam generator supports, reactor vessel internals (RVI), pressurizer, control element drive mechanisms (CEDMs), head lift rig (HLR), in core instrumentation (ICI) tubes, Reactor Vessel Gas Vent System (RVGVS), and fuel and control element assemblies (CEAs).
(2)Maintain the capability to insert the control element assemblies (CEAs).

The related UFSAR-described methods of evaluation are:

(1)UFSAR Section 3.9.1.2.2.1.11 describes ANSYS as a large-scale, general-purpose, finite element program for linear and nonlinear structural and thermal analysis of reactor coolant loop components.
(2)Seismic analysis of the reactor internals is discussed in UFSAR Section 3.7.3.14 and 3.7.3.14.1, which references Combustion Engineering Topical Report CENPD-178.

Subsection (b) ECCS Performance

The design functions of the ECCS are to mitigate the effects of a large break loss of coolant accident (LBLOCA) and a small break loss of coolant accident (SBLOCA) and to maintain post-LOCA long term cooling to meet the acceptance criteria of 10 CFR 50.46 [UFSAR 6.3.3].

The methods of evaluation for ECCS performance analyses are explicitly described in UFSAR 6.3.3.2 (LBLOCA effects), 6.3.3.3 (SBLOCA effects), and 6.3.3.4 (post-LOCA long term cooling), which includes a citation to NRC-approved computer programs and analytical methods documented in Combustion Engineering Topical Report CENPD-254-P-A.

Subsection (c) Non-LOCA Transients

The design functions of the steam generators and the systems, structures and components (SSCs) credited for mitigating the effects of the non-LOCA transients, per UFSAR Sections 5.4.2.1 and 5.2, are:
(1)Transfer heat from the primary to the secondary system as required for plant modes in which the steam generators must be operable, and
(2)Provide a part of the reactor coolant pressure boundary.

The UFSAR-described method of evaluation for non-LOCA transient events utilizes the CESEC computer program for loss of all normal feedwater flow (UFSAR 15.10.2.2.5) and loss of all feedwater flow with a concurrent single failure of an active component (UFSAR 15.10.2.3.2).
Subsection (d) Containment P-T Analysis

The design functions relevant to this subsection are:
(1)UFSAR Table 6.2-3 states that the containment internal design pressure is 60.0 psig and the containment design temperature is 300°F. Technical Specification (TS) Bases B3.6.5 clarifies that the temperature limit of 300°F is not an ambient atmosphere temperature limit, but rather pertains to the containment structure such as the containment liner plate and concrete. Per UFSAR Section 6.2.2.1.3, the design temperature of equipment

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subjected to LOCA or MSLB conditions is 300°F. Per UFSAR Table 6.2-34 the containment emergency cooling unit (ECU) atmosphere inlet temperature is limited to 300°F.
(2)UFSAR Sections 6.2.1.1.1.4, 6.2.2.1.1C and 6.2.2.2.1A, state that the containment emergency fan cooler system, in conjunction with the Containment Spray System (CSS) and the shutdown heat exchangers (i.e., one train of each system), is capable of reducing the post-LOCA containment pressure from the peak value to one-half of the peak value in 24 hours in accordance with 10 CFR 50, Appendix A, GDC 38 and Standard Review Plan 6.2.1.1.A.

The UFSAR-described methods of evaluation are:
(1)UFSAR Section 6.2.1.1.3.1.C states that the containment pressure analyses are performed using the COPATTA computer program and specifically describes some aspects of the COPATTA modeling.
(2)UFSAR Section 6.2.1.3.3 states that the LOCA blowdown phase is simulated with the CEFLASH 4A computer program and specifically describes some aspects of the CEFLASH 4A modeling.
(3)UFSAR Section 6.2.1.3.4 states that the LOCA core reflood and post-reflood phase is simulated with the FLOOD3 computer program and specifically describes some aspects of the FLOOD3 modeling.
(4)UFSAR Section 6.2.1.3.5 states that the LOCA long term phase M-E release is analyzed with the use of the COPATTA and CONTRANS computer programs and specifically describes some aspects of the decay heat model.
(5)UFSAR Section 6.2.1.4 states that the MSLB M-E data is analyzed with the use of the SGNIII computer program and UFSAR Section 6.2.1.4.4 specifically describes some aspects of the SGNIII modeling.
(6)UFSAR Section 6.2.1.4.3 states that a 2 percent increase in the initial inventory resulting from thermal and pressure expansion of the steam generator at operating conditions is to be included in the analysis.
(7)Various locations within UFSAR Section 6.2 include references to the M-E and peak P-T analysis power level being 3390 MWt plus 2 percent uncertainty (although the plant now operates at 3458 MWt).

Subsection (e) Low Temperature Overpressure Protection (LTOP)

The design function relevant to this subsection is the ability of the shutdown cooling system (SDCS) relief valve to terminate any inadvertent pressure transient occurring when the RCS temperatures are below the applicable P-T operating curve temperatures, corresponding to pressures below the relief valve setpoint pressure, as described in UFSAR, Section 5.2.2.11.2.2. Above the LTOP enable temperature, overpressure protection is provided by the pressurizer safety valves (as the SDCS relief valve is isolated from the RCS).

The UFSAR describes mass and energy addition transients analyzed to demonstrate acceptable LTOP for the existing plant configuration. The methods used to analyze the LTOP transients are not described in the UFSAR.

Subsection (f) RPS, ESFAS, COLSS and CPC

The design functions pertinent to this subsection are:
(1)The Reactor Protective System (RPS) trips the reactor to protect the core fuel design limits and reactor coolant system pressure boundary for anticipated operational occurrences, and also assist in limiting conditions for certain accidents (UFSAR Section 7.2.1.1).
(2)The Engineered Safety Features Actuation System (ESFAS) generates signals that actuate the Engineered

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Safety Features (ESF) systems (UFSAR Section 7.3).
(3)The Core Operating Limit Supervisory System (COLSS) monitors the limiting conditions for operation on peak linear heat rate, margin to departure from nucleate boiling, total core power, axial shape index and azimuthal tilt (UFSAR Section 7.7.1.5.1).
(4)The Core Protection Calculators (CPCs) calculate departure from nucleate boiling ratio and linear power density and initiate a trip on those parameters (UFSAR Section 7.2.1.1.2.5).

UFSAR Section 7.2.1.2, "Design Bases," states: "A detailed description of the methodology used to calculate the setpoints, including their explicit uncertainties, is provided in the report CEN 112(S) Revision 0, Plant Protection System Selection of Trip Setpoint Values, November 15, 1979."

Subsection (g) NSSS Performance

The design functions of the affected systems are:
(1)The RCS circulates water in a closed cycle, removing heat from the reactor core and internals and transferring it to the secondary (steam generating) system (UFSAR Section 5.1).
(2)The CVCS maintains the chemistry, purity and activity level of the reactor coolant within limits, maintains the required volume of water in the RCS, provides a controlled discharge path for reactor coolant to the radwaste system, controls RCS boron concentration, provides auxiliary pressurizer spray, provides means for testing the check valves that isolate the safety injection system from the RCS, provides continuous measurement of reactor coolant boron concentration, collects reactor coolant pump seal controlled bleed-off, provides for leak testing the RCS, provides a normal and an alternate means for filling the RCS (UFSAR Section 9.3.4.1.1).
(3)In the event of a LOCA, the SIS provides core cooling sufficient to prevent significant alteration of core geometry, precludes fuel melting, limits the cladding metal water reaction, removes energy generated in the core for an extended period of time and maintains the core sub-critical for the duration of the LOCA (UFSAR Section 6.3.1).
(4)The SDCS removes heat from the RCS during post-shutdown periods (UFSAR Section 5.4.7.1).
(5)The CSS removes heat from the containment atmosphere in the event of a LOCA or MSLB inside containment (UFSAR Section 6.2.2.1).

The methods of evaluation for NSSS performance analyses are not described in the UFSAR.

Subsection (h) Non-Safety Related Control Systems Performance

The design functions of the affected systems are:
(1)The Pressurizer Pressure Control System (PPCS) maintains pressurizer pressure within specified limits (UFSAR Section 7.7.1.2.1).
(2)The Pressurizer Level Control System (PLCS) regulates Reactor Coolant System (RCS) water inventory (UFSAR Section 7.7.1.2.2).
(3)The Feedwater Control System (FWCS) maintains steam generator water level within acceptable limits (UFSAR Section 7.7.1.3).
(4)The Steam Bypass Control System (SBCS) provides a means for controlling NSSS thermal energy during plant startup, cooldown and hot standby, and accommodates load rejections, unit trips and other conditions that result in generation of excess energy by the NSSS (UFSAR Section 7.7.1.4.1).

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The Methods of Evaluation for the non-safety control system performance are not described in the UFSAR.

SECTION II: 50.59 SCREEN CONCERNS (See Keypoints Section II)

State if the proposed activity:

1. Changes an SSC in a manner that adversely affects the UFSAR/DSAR design function(s) or has an adverse affect on the method of performing or controlling UFSAR/DSAR design function(s).

No.
Replacement of the OSGs with the RSGs is a replacement in-kind in terms of form, fit, and function, and is designed to involve no, or only minimal, permanent modifications to the plant structures, systems or components. In fact, installation of the RSG requires only minor modifications to the blowdown piping routing, instrument sensing line routing, access platform configuration and possibly steam generator key bracket configuration.

The RSGs are designed and fabricated as ASME Section III, Class 1 components in accordance with the requirements of the newer ASME Code edition than that used to design and fabricate the OSGs. During fabrication, the steam generator parts and assemblies were non-destructively examined and tested in accordance with the ASME Code, Section III requirements. The steam generators were also subject to additional inspections and tests beyond the Section III requirements. These included ultrasonic testing of tubesheet cladding for defects, ultrasonic testing of weld cladding for bond integrity, and helium leak rate testing of the tube-to-tubesheet welds.

Shop hydrostatic tests of the primary and secondary side of the steam generators were conducted in accordance with the ASME Code, Section III, and as replacement components the RSG were subject to pre-service examinations in accordance with ASME Code, Section XI requirements.

The fact that the RSG were designed, fabricated and examined to the newer edition of the Code is an enhancement over the OSGs.

The RSGs are designed to perform the same design functions as those currently performed by the OSGs (see Section I.3), and the differences listed in Section I.2 do not affect their ability to perform these design functions. Specifically, installation of the RSGs does not require any changes to the steam generator water level, Feedwater Control System (FWCS), ADV or MSSV setpoints.

The values of the RSG major design parameters are different than the values of the corresponding OSG parameters (see Section I.2). The RSG steam flow is slightly higher, the outlet steam pressure is lower and the moisture content is considerably lower than the values for the OSGs. These changes are in a conservative direction (pressure) or constitute an improvement over the OSGs (moisture content).

The RSG design parameters are achieved at the same feedwater and reactor coolant inlet (Thot) temperatures as the current OSG operating temperatures, with an approximately the same NSSS reactor coolant flow rate as the rate currently measured with the OSGs installed. However, it is achieved with a considerably larger heat

Operation Details Cont'd:

transfer surface area.

The RSG heat transfer area is larger than the OSG area ([ ] ft2 vs. ~ [ ] ft2) and the RSG tube bundle is taller than the OSG bundle. The effects of the larger tube bundle are documented in Parts 3 and 4 of Reference 1. The larger and taller RSG tube bundle offers an increased driving head for natural circulation flow, so that the RSGs will perform the same as, or better than, the OSGs during the events that credit natural circulation. The steam flow limiter integral to the RSG steam outlet nozzle does not impact natural circulation cooldown, as the Atmospheric Dump Valve (ADV) discharge port size is limiting for RCS cooldown.

The RSG primary side volume is larger than the OSG volume (2003 ft3 vs. 1895 ft3). Due to this increase, the time necessary to cooldown and depressurize the RCS to Shutdown Cooling System (SDCS) entry conditions (approximately 350oF and 376 psia) will be extended slightly. This time increase is insignificant when compared to the overall cooldown time and the available Technical Specification inventory in the Condensate Storage Tanks.

Likewise, the RCS volume increase results in an insignificant time extension for the SDCS to reduce RCS temperature. Specifically, it will take an additional 1.2 minutes to reach a refueling temperature of 130oF for dual-train SDCS operation (4.2 minutes for single-train operation). Cold shutdown is defined as the RCS temperature being 200oF, which takes much less time than cooling to 130oF. This slight time increase does not impact compliance with Branch Technical Position (BTP) Reactor Systems Branch (RSB) 5-1 (i.e., achieving cold shutdown conditions within 24 hours from inside the control room with a concurrent single failure). Additionally, the time increase does not adversely affect the ability to achieve and maintain post-fire safe shutdown (cold shutdown) within 72 hours following the fire (10CFR50, Appendix R, III.L), because this time increase is negligible compared to the shutdown time frame in the RSB 5-1 and Appendix R events.

The RCS volume increase will also result in a slight increase of the containment flooding level, following a LOCA. This increase was evaluated from the perspective of an impact on the instruments required for post-LOCA response and it was concluded that the change in the flooding level was acceptable (Reference Calc. N-4060-030, ECN A54375).

The RSG tube wall is thinner than the OSG tube wall (0.0429 in. vs. 0.048 in.). Regulatory Guide 1.121 analysis of the RSG tubes was performed consistent with the loading conditions and acceptance criteria guidance provided in the guide and includes assessment of the loads that result from postulated accident conditions, including a design basis earthquake (DBE) in combination with a LOCA and MSLB. The analysis concluded that a tube would have to be plugged if it contained a flaw to a depth lesser than that for the OSGs (35% vs. 44%). This reduction of the tube plugging limit is conservative.

The RSG major materials of construction are different than the OSG materials. These differences represent a vast improvement over the OSG materials in terms of corrosion, erosion-corrosion and wear resistance. The RSG primary and secondary side pressure boundary materials have nil-ductility (ndt) reference temperature (RT) the same as or better than those for the Unit 2 OSG materials (70oF limit in Unit 2 LCS 3.7.109). The Unit 2 OSG limitation of 70oF is based on an RTndt of [ ] oF, whereas the RSG limitation of 70oF is based on an RTndt of [] oF (ref. Unit 2 LCS 3.7.109). Lowering the RTndt reflects a corresponding improvement in material properties which is not adverse.

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A change was also made to LCS 3.7.109 to recognize that periodic surveillance of compliance to the above secondary side temperature limit is not required during the time that the OSGs are drained for removal from the plant (with the core off-loaded) and the time the RSGs have been installed and may be pressurized (prior to reloading fuel in the core). This administrative LCS change is not adverse, since the OSGs and RSGs do not need to be operable during this SG replacement time frame.

The configuration and/or number of the RSG internal components (tube bundle, feedwater ring, moisture separators, steam dryers) are different than those for the OSG components. These differences also represent functional improvements over the OSG components. The RSGs have a different number of access provision then the OSGs, some of which have also different dimensions. These differences are aimed at improving RSG accessibility for inspections and maintenance.

The RSG lower shell diameter is slightly larger than the OSG diameter (174.6 in. vs. [ ] in.) and the RSG are slightly heavier than the OSGs. However, these differences are within the allowable limits established and allowed for the RSGs.

Based on the above, it is concluded that, the proposed activity does not adversely affect the steam generator ability to:
(1)Function as a part of the RCPB
(2)Transfer heat between RCS and main steam system and
(3)Remove heat from the RCS to achieve and maintain safe shutdown following postulated accidents (other than the large break LOCA).

Based on their physical characteristics, the RSGs provide specific heat transfer capability under defined plant conditions and provide RCPB integrity under defined limits of pressure and temperature. However, replacing OSGs with RSGs does not directly affect any methods of performing or controlling heat transfer or RCPB integrity. The indirect effects of the steam generator replacement on the methods of performing or controlling design functions of other SSCs was evaluated in Sections 3.1 through 3.12 of Appendix A of Reference 1 and it was concluded that the replacement did not adversely affect the ability of performing or controlling any of these design functions. Detailed discussion of these aspects of the replacement is provided in the following paragraphs.

Subsection (a) RCS Structural Integrity

No. Implementation of steam generator replacement required evaluations and analyses accounting for dead weight, normal operating pressure, seismic, and branch line pipe break (BLPB) loads to demonstrate that structural integrity of the evaluated components was maintained for operation with the RSGs installed. The analyses and evaluations concluded that structural integrity would be maintained.

The number of allowed opening and closing cycles for the primary manway nuts and stud bolts was decreased from 500 to 200. Based on SONGS operating experience over the last 25 years, it was determined that such reduction did not constitute an adverse change. however, as the number of manway opening/closing cycles is not described in the UFSAR, and the design meets the ASME Code criteria.

Subsection (b) ECCS Performance

Operation Details Cont'd:

No. Reference 2 documents evaluation of the differences between the RSGs and OSGs which showed that most of the differences are small in magnitude and/or have a negligible impact on ECCS performance. Where the differences are larger or are associated with parameters that have a significant impact on ECCS performance, the differences were found to have an overall beneficial effect on ECCS performance.

Subsection (c) Non-LOCA Transients

No. The results of analyses and evaluations of UFSAR Chapter 15 non-LOCA transients performed for operation with RSGs are presented in Part 3, Section 3.7 of Reference 1. Reanalyzed events continue to meet acceptance criteria when accounting for RSG characteristics with the performance of mitigating SSCs within Technical Specification limits. For evaluated events, the existing analyses were shown to remain applicable. Events classified as "Not Reanalyzed" are not affected by the RSGs, and events that are "Not Applicable" are not relevant to the non-LOCA transient analyses.

Subsection (d) Containment P-T Analysis

No. The containment P-T analysis was revised to account for the RSGs. The governing case is the MSLB with MSIV failure, for which peak containment pressure of 56.5 psig for the OSGs decreased to 51.5 psig for the RSGs. The MSLB with MSIV failure peak containment vapor temperature of 409°F for the OSGs decreased to 380°F for the RSGs. Note that the data for the governing case are for the worst case MSLB with MSIV failure case, which changed from the full power case for the OSGs to the zero power case for the RSGs. Although the peak LOCA containment pressure increased from 45.9 psi with the OSGs to 46.7 psi with the RSGs, the peak pressure for the RSGs remains below that for the limiting event (MSLB), which is also below the containment design pressure of 60 psig (see above).

The pressure profiles at 24 hours post-LOCA show that the pressure is below one-half of the peak pressure, thereby confirming compliance with GDC 38. The MSLB event is analyzed for 5000 seconds (1.39 hours) at which time the pressure is well below one-half of the peak pressure, thereby confirming compliance with GDC 38 for MSLB as well. As such, SG replacement is not adverse from the containment P-T analysis perspective.

Subsection (e) Low Temperature Overpressure Protection

No. Reanalysis of the LTOP transients demonstrated that the existing RCS Technical Specification pressure-temperature limits are met and that the integrity of the RCPB is maintained. The SDCS peak pressure does not exceed the ASME Code allowable. In particular, the reanalysis used the same initial conditions and SDCS relief valve characteristics as described in the UFSAR and analyzed the same LTOP transients. The reanalysis demonstrated that the relieving capacity of the SDCS relief valve remains adequate. Since there were no design changes to the LTOP mitigation equipment, no changes to the LTOP administrative controls, and no changes to operator actions, SG replacement has no effect on how design functions that provide LTOP are accomplished or controlled.

Subsection (f) RPS, ESFAS, COLSS and CPC

Operation Details Cont'd:

No. SG replacement is independent of, and does not alter, the RPS, ESFAS, COLSS or CPCs, or change the manner in which RPS, ESFAS, COLSS or CPC design functions are performed or controlled.

Subsection (g) NSSS Performance

No. For SG replacement, functional and performance requirements for the RCS, CVCS, SIS, SDCS and CSS are met based on existing SSC capabilities. Section 3.10 of Reference 1 addresses that the RCS, CVCS, SIS, SDCS, and CSS will continue to perform their design functions with the RSGs installed. SG replacement does not change the manner in which design functions of the RCS, CVCS, SIS, SDCS and CSS are performed or controlled.

Subsection (h) Non-Safety Related Control Systems Performance

No. The non-safety Control System configuration and setpoints currently used for SONGS Unit 2 are adequate for operation with the RSGs installed. These control systems ensure proper NSSS dynamic response such that the probability for reactor trips is minimized for the analyzed load changes and certain equipment malfunctions, and so that the post-trip responses are acceptable. There are no changes to control systems design or to automatic or manual control capability, so there is no effect on how design functions of these systems are performed or controlled.

Conclusion: Based on the foregoing, changing the OSGs to RSGs does not change an SSC in a manner that adversely affects UFSAR-described design functions or that has an adverse effect on the method of performing or controlling UFSAR-described design functions.

2. Changes a procedure in a manner that adversely affects how UFSAR/DSAR described SSC design function(s) are performed or controlled.

No.
The RSGs provide RCPB integrity under defined limits of pressure and temperature and provide specific heat transfer capability under defined plant conditions based only on their physical characteristics. Procedures will be updated, as necessary, to reflect the RSGs and the updated analyses. The analyses have shown that the RCS and other systems will operate in a manner such that they will continue to perform their design functions. Therefore, steam generator replacement does not involve changing any procedure in a manner that adversely affects how UFSAR-described design functions related to the steam generators are performed or controlled. The effect of the steam generator replacement on procedures pertinent to other SSCs was evaluated in Sections 3.1 through 3.12 of Appendix A of Reference 1. Changes to procedures required to reflect the RSGs will not adversely affect the methods of performing or controlling UFSAR-described functions.

3. Involves revising or replacing a UFSAR/DSAR described method of evaluation that is used in establishing the design bases or used in the safety analyses in an adverse manner.

Operation Details Cont'd:

Yes.
The stress analyses for the RSGs utilized the ABAQUS computer program, whereas UFSAR Sections 3.9.1.2.2.1.11 and 3.9.1.2.2.2.3 describe these analyses for the OSGs as utilizing the ANSYS computer program.

The RG 1.121 tube wall thinning evaluation for the RSGs changed the methods of evaluation from those described in UFSAR 5.4.2.3.1.3. Instead of using the CEFLASH computer program for MSLB mass-energy release analysis, the peak forces acting on the RSG tubes were calculated manually using the maximum possible differential pressure across the tube wall. In addition, instead of using a two-step process with the STRUDL and ANSYS computer programs to calculate displacement histories and tube stresses, respectively, the tube stresses were determined from blowdown forces using only the ANSYS computer program. Finally, for the RSGs, the tube stresses due to LOCA, DBE and MSLB were combined into one "limiting event" and the stresses for this combined event were calculated, whereas for the OSGs, the analyses considered primary loop branch line pipe break plus DBE and MSLB plus DBE separately.

Subsection (a) RCS Structural Integrity

No. The original RCS analysis to generate seismic response spectra for RCS branch nozzle and subsystem analysis was performed using the STRUDL computer code to define the structural properties of the RCS coupled to the containment structure, basement and soil. More recent analyses performed in support of Tcold reduction, inoperable steam generator snubbers during Modes 4 and 5, as well as those for steam generator sensitivity studies, used the ANSYS computer code to define the structural properties of the RCS. The equivalence of the two models (ANSYS and STRUDL) was confirmed by comparison of their frequencies and mode shapes. The ANSYS code is an approved code for RCS structural analysis at SONGS, therefore, the change in the method of evaluation from STRUDL to ANSYS is not adverse.

Yes. The seismic analysis of the RVI was performed in accordance with the methodology described in topical report CENPD-178-P, Revision 1-P. UFSAR Section 3.7.3.14.1 references topical report CENPD-178. CENPD-178 was originally submitted to the NRC in 1976 as a Combustion Engineering topical report. Subsequent to submittal of CENPD-178, modeling techniques, computer codes, testing methods and acceptance criteria were modified in response to changes in licensing requirements. As a result, the entire report was revised and resubmitted as CENPD-178-P, Revision 1-P. The use of CENPD-178-P, Revision 1-P, is a revision to the UFSAR-described method of evaluation used in establishing the design bases.

Subsection (b) ECCS Performance

No. Instead of reanalyzing the UFSAR-described ECCS performance evaluations for LBLOCA, SBLOCA, and post-LOCA long term cooling, an evaluation was performed to assess the effects of revised parameters on the Analyses of Record (AORs) to determine whether the AORs remain valid. This evaluation is documented in Reference 2. The AOR methodologies were not revised or replaced.

Subsection (c) Non-LOCA Transients

No. The transient simulation code used for analyzing the loss of all normal feedwater flow and loss of all feedwater flow with a concurrent single failure of an active component was changed from the CESEC

Operation Details Cont'd:

computer program (for OSGs) to the CENTS computer program. However, this change is consistent with Reference 3, which approved use of the CENTS computer program for non-LOCA transient analysis for SONGS Units 2 and 3. The CENTS computer program is being used within the constraints and limitations associated with the method; therefore, the change in method of evaluation is not adverse. No other non-LOCA transient methods of evaluation were revised or replaced.

Subsection (d) Containment P-T Analysis

No. For the RSGs, the blowdown phase of the LOCA mass and energy release was simulated with the CEFLASH 4A computer program consistent with UFSAR Section 6.2.1.3.3. The reflood and post-reflood phases of the LOCA for cold leg breaks were simulated using the FLOOD3 computer code consistent with UFSAR Section 6.2.1.3.4. The long term cooling mass and energy releases were determined with the CONTRANS and COPATTA computer programs. CONTRANS calculated residual heat addition from the primary and secondary metal and SG inventory. The residual heat addition was input into COPATTA. Decay heat was developed consistent with the prior analysis and input directly into COPATTA. This is the same methodology as that described in UFSAR Section 6.2.1.3.5.

Subsequent to the original analysis for the OSGs, the NRC granted SONGS Units 2 and 3 a change in the licensed power limit and an exception from the 10 CFR Part 50, Appendix K, requirement to use a 2 percent power measurement uncertainty in ECCS/LOCA related analyses. The sum of the licensed power limit and the power measurement uncertainty did not change as a result of the NRC action. Consequently, the "analysis power" (licensed power plus power measurement uncertainty) for the decay heat model, as well as the full power M-E and P-T cases, did not change even though the NRC granted an apparent change in the licensed full power level.

For the RSGs, the MSLB containment M-E analysis was performed using the SGNIII computer program, consistent with UFSAR Sections 6.2.1.4 and 6.2.1.4.4. A 2 percent expansion factor was used which is consistent with UFSAR Section 6.2.1.4.3. The mass and energy data was then used to determine the containment response using the COPATTA computer program. This is the same methodology as that described in UFSAR Section 6.2.1.1.3.1.C.

Based on the above, containment P-T re-analyses were performed in accordance with UFSAR-described methods of evaluation.

Subsection (e) Low Temperature Overpressure Protection

No. The UFSAR does not describe a method of evaluation for analyzing LTOP transients.

Subsection (f) RPS, ESFAS, COLSS and CPC

No. The methodology described in report CEN 112(S), Rev. 0, cited in UFSAR Section 7.2.11.2 for calculating setpoints, including uncertainties, was used in evaluating RSG effects on setpoints and uncertainties.

Operation Details Cont'd:

Subsection (g) NSSS Performance

No. RSG effects on the NSSS systems were assessed using qualitative evaluations, manual calculations and the results of various analyses and evaluations. These evaluations, calculations and analyses do not involve revising or replacing a UFSAR-described method of evaluation that was used in establishing the design bases or used in the safety analyses.

Subsection (h) Non-Safety Related Control Systems Performance

No. The methods of evaluation for non-safety control system performance are not described in the UFSAR.

Based on the above, it is concluded that the computer programs used for stress analysis of the RSGs and for seismic analyses of the reactor vessel internals were changed. These changes constitute a revision to the methods of evaluation described in UFSAR 3.7.3.14.1, 3.9.1.2.2.1.11, 3.9.1.2.2.2.3 and 5.4.2.3.1.3..

4. Involves a test or experiment not described in the UFSAR/DSAR, where an SSC is utilized or controlled in a manner that is outside the reference bounds of the design for that SSC, or is inconsistent with the analysis or descriptions in the UFSAR/DSAR.

No.
The proposed activity involves functional testing of the RSGs after installation. The testing will be performed during normal operation of the plant when the RSGs and the associated SSCs will be utilized strictly within the bounds of their design and consistent with the approved plant operating procedures. The testing will involve only recording of the selected plant operating parameters in accordance with the approved test procedure, using the instruments installed in the plant. The effect of RSG installation on requiring tests and experiments involving other SSCs was evaluated in Sections 3.1 through 3.12 of Appendix A of Reference 1. The RSGs do not require any tests or experiments not described in the UFSAR, where an SSC would be utilized or controlled in a manner that would be outside the bounds of the design, or inconsistent with the analyses of record for that SSC.

5. Requires a change to the Technical Specifications.

Yes.
Steam generator replacement results in a change in tubing material (from Alloy 600MA to Alloy 690TT) and wall thickness (from 0.048 in. to 0.0429 in.). The new tubing was evaluated in accordance with Regulatory Guide 1.121 in order to determine new tube plugging criteria. For OSGs, Technical Specification 5.5.2.11 requires that tubes be plugged or repaired if in-service inspection determines that the tube contains flaws to a depth equal to, or greater than, 44% of the nominal tube wall thickness. The RSG tubes will be required to be plugged if they contain flaws to a depth equal to, or greater than, 35% of the nominal wall thickness.

Operation Details Cont'd:

Therefore, the proposed activity involves changes to Technical Specifications 3.4.17, "SG Tube Integrity," 5.5.2.11, "SG Program," and 5.7.2, "Special Reports."

Subsection (a) RCS Structural Integrity

No. SG replacement required assessing its effect on RCS components. The assessment concluded that no Technical Specifications were affected.

Subsection (b) ECCS Performance

No. The Technical Specifications that apply to the ECCS system and analyses, 3.5, "Emergency Core Cooling Systems," and 5.7.1.5, "Core Operating Limits Report (COLR)," were reviewed for impact and do not require changes.

Subsection (c) Non-LOCA Transients

No. Review of the Technical Specifications revealed one change required for the non-LOCA transient analyses and evaluations. License amendments 212 (Unit 2) and 204 (Unit 3) established revised high power level trip values in Technical Specification 3.7.1 for three and four MSSVs out of service per steam generator. The revised trip values are consistent with the values documented in Part 3, Section 3.7 of Reference 1. Although revised Technical Specification 3.7.1 also increases the allowable power level for one and two MSSVs inoperable per steam generator, this aspect of the Technical Specification change is not required by and will not be impacted by the RSGs. Hence, SG replacement does not require any Technical Specification changes for non-LOCA transients.

Subsection (d) Containment P-T Analysis

Yes. As discussed above in II.1 (d), the peak containment pressure of 56.5 psig for the OSGs decreases to 51.5 psig for the RSGs. Therefore, the proposed activity does involve a change to Technical Specification 5.5.2.15, "Containment Leakage Rate Test Program," (and its corresponding Bases) and LCS 3.6.100, "Pre-Stressed Concrete Containment Tendon Surveillance Program."

Subsection (e) Low Temperature Overpressure Protection

No. SG replacement required reanalysis of the LTOP mass and energy transients but no Technical Specification changes.

Subsection (f) RPS, ESFAS, COLSS and CPC

No. SG replacement does not involve any changes to the Technical Specifications pertinent to the RPS, ESFAS, COLSS or CPCs.

Subsection (g) NSSS Performance

Operation Details Cont'd:

No. SG replacement does not involve any changes to the Technical Specifications pertinent to the RCS, CVCS, SIS, SCS, or CSS.

Subsection (h) Non-Safety Related Control Systems Performance

No. The Technical Specifications do not include non-safety control system setpoints or performance requirements. As a result, the proposed activity does not require changes to any Technical Specifications.

Based on the above, it was concluded that the proposed activity involves changes to Technical Specification 3.4.17, "Steam Generator (SG) Tube Integrity," 5.5.2.11, "Steam Generator Tube Surveillance Program," 5.7.2, "Special Reports" and TS 5.5.2.15, "Containment Leakage Rate Testing Program," as well as to LCS 3.7.109, "Steam Generator Pressure/Temperature Limitation".

The aspect of the proposed activity requiring changes to the Technical specifications are addressed in PCN-583 submitted for NRC approval..

SECTION III: SCREEN CONCLUSION (See Keypoints Section III)

Based on the responses provided in Section II, the proposed activity is:

____ NOT adverse

__X__ Adverse, record the 50.59 Evaluation (SE) notification and task: 800175663-0520

The 50.59 screen concern that requires evaluation is Screen Concern 3. Refer to Section II.3 and Subsection II..3(a) for further information.

Specifically, the following issues have to be evaluated:

1. The OSG structural integrity analyses described in UFSAR Sections 3.9.1.2.2.1.11 and 3.9.1.2.2.2.3 utilized the ANSYS computer program, whereas the RSG analyses utilized the ABAQUS computer program.

2. The seismic analysis of the RVI for the RSGs was performed in accordance with the methodology described in topical report CENPD-178-P, Revision 1-P. The OSG analyses described in UFSAR Section 3.7.3.14.1 reference CENPD-178.

3. The RG 1.121 tube wall thinning evaluation for the RSGs used methods of evaluation different from those described in UFSAR 5.4.2.3.1.3.

(a)The OSGs used the computer program CEFLASH for MSLB mass-energy release analysis, the RSG analyses used manual calculation methods using the maximum differential pressure across the tube wall.
(b)Instead of using a two-step process with the computer programs STRUDL and ANSYS to calculate displacement histories and tube stresses, respectively, tube stresses were determined from blowdown forces using only ANSYS.

Operation Details Cont'd:

(c)For the RSGs, the LOCA, DBE and MSLB events were combined into one "limiting event" and the stresses for this combined event were calculated, whereas the OSG analyses considered LOCA plus DBE [UFSAR 5.4.2.3.1.3.A] and MSLB plus DBE [UFSAR 5.4.2.3.1.3.B] as separate events.

Technical Specification changes are required as a result of the proposed activity. The Technical Specification changes identified in this screen are included in PCN-583 sent for NRCs approval, and are hence excluded from the 50.59 Evaluation of the proposed activity. NRC approval of PCN-583 is a prerequisite to closing the NECP governing this proposed activity

Updated Final Safety Evaluation Report (UFSAR) changes are required as a result of the proposed activity. Specifically, the UFSAR sections listed below are affected by the proposed activity and are revised accordingly. Prior NRC approval is either being sought through PCN-583, or is not required as demonstrated in this screen. The changes are "Pending" NRC approval of PCN-583 and subsequent closure of NECP 800071702 (Unit 2 RSG) and 800071703 (Unit 3 RSG).

3.7	Seismic Design	change # 2008-38
3.9	Mechanical Systems and Components	change # 2008-10
3.11	Environmental Design (P-T Analysis)	change # 2008-11
4.4	Thermal and Hydraulic Design	change # 2008-12
5.1	Summary description	change # 2008-13
5.2	Integrity of RC Pressure Boundary	change # 2008-14
5.4	Component and Subsystem Design	change # 2008-15
6.2	Containment systems	change # 2008-16
6.3	Emergency Core Cooling System	change # 2008-17
7.2	Reactor Protective System	change # 2008-18
9.3	Process Auxiliaries	change # 2008-19
10.1	Summary description	change # 2008-20
10.3	Main Steam Supply System	change # 2008-21
15.10	Updated accident Analyses	change # 2008-33

SECTION IV: REFERENCES (See Keypoints Section IV)

1.WCAP-16811-P, "SONGS Units 2 and 3 Replacement Steam Generator Project, NSSS Licensing Topical Report."
2.CN-OA-05-71, "Evaluation of Impact of RSGs on the SONGS Units 2 and 3 ECCS Performance Analysis."
3.Letter from H.N. Berkow (NRC) to G. Bischoff (Westinghouse Electric Company), November 24, 2004, Subject: Final Safety Evaluation for Topical Report WCAP-15996-P, "Technical Description Manual for CENTS Code."

UFSAR Sections Reviewed:

1	Introduction and General Description of Plant
1.2.3	Nuclear Steam Supply System (NSSS)
3.	Design of Structures, Components, Equipment and Systems
3.7	Seismic Design
3.9.	Mechanical Systems and Components

Operation Details Cont'd:

App. 3.9A Dynamic System Analyses of Reactor Coolant System, Reactor Internals ECCS Piping, and CEDMs Under Faulted Conditions
3.11 Environmental Design of Mechanical and Electrical Equipment
4 Reactor
4.4 Thermal and Hydraulic Design
5 Reactor Coolant and Connected Systems
5.1 Summary Description
5.2 Integrity of Reactor Coolant Pressure Boundary (RCPB)
5.4. Component and Subsystem Design
6 Engineered Safety features
6.2. Containment Systems
6.3 Emergency Core Cooling System
7 Instrumentation and Controls
7.2 Reactor Protective System
7.3 Engineered Safety Features System
7.7 Control Systems not Required for Safety
8 Electric Power
8.3 Onsite Power Systems
9 Auxiliary Systems
9.3. Process Auxiliaries
10 Steam and Power Conversion Systems
10.1 Summary Description
10.3 Main Steam Supply System
10.4 Other Features of Steam and Power Conversion System
15 Accident Analyses
15.6 Decrease in Reactor Coolant Inventory
15.10 Updated Accident Analyses
App. 15.D Responses to IE Bulletins 79 27 and 80 06, IE Information Notice 79 22, and Multiple Control System Failure Evaluation

Technical Specifications Reviewed:
3.1.9 Boration Systems Operating
3.1.10 Boration Systems Shutdown
3.3.1 Reactor Protective System (RPS) Instrumentation Operating
3.3.2 Reactor Protective System (RPS) Instrumentation Shutdown
3.3.5 Engineered Safety Features Actuation System (ESFAS) Instrumentation
3.4 Reactor Coolant System
3.4.3 RCS Pressure and Temperature (P/T) Limits
3.4.4 RCS Loops Modes 1 and 2
3.4.5 RCS Loops Mode 3
3.4.6 RCS Loops Mode 4
3.4.7 RCS Loops Mode 5, Loops Filled
3.4.12.1 Low Temperature Overpressure Protection (LTOP) System, RCS Temperature d PTLR Limit
3.4.12.2 Low Temperature Overpressure Protection (LTOP) System, RCS Temperature >

Order: NECP **800175663**
Description: **Steam Generator Replacement Mstr ECP U2**
Operation No: 0530 **Sub Op:** **Op Wrkctr:** ED_SGD SGRP Design

Operation Details Cont'd:

PTLR Limit

3.4.13	RCS Operational Leakage
3.4.17	RCS Steam Generator (SG) Tube Integrity
3.5	Emergency Core Cooling Systems (ECCS)
3.6.6.1	Containment Spray and Cooling Systems
3.7.1	Main Steam Safety Valves (MSSVs)
3.7.2	Main Steam Isolation Valves (MSIVs)
3.7.3	Main Feedwater Isolation Valves (MFIVs)
3.7.5	Auxiliary Feedwater (AFW) System
3.8.1	AC Sources Operating
3.9.4	Shutdown Cooling (SDC) and Coolant Circulation High Water Level
3.9.5	Shutdown Cooling (SDC) and Coolant Circulation Low Water Level
5.5.2.11	Steam Generator (SG) Program
5.5.2.15	Containment ILRT
5.7.1.5	Core Operating Limits Report (COLR)
5.7.1.6	Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR)

Licensee Controlled Specifications Reviewed:
3.7.109 Steam Generator Pressure/Temperature Limitation

SECTION V: SUMMARY (See Keypoints Section V)
This 10CFR50.59 Screen operation replaces operation 800175663-0510 with the following corrections.

1. End of Section II: Changed to LCS 3.7.109, "Steam Generator Pressure/Temperature Limitation"

instead of LCS 3.6.100, "Pre-stressed Concrete Containment Tendon Surveillance Program".

2. Beginning of Section III; The operation of the 50.59 Evaluation (SE) should be 800175663-0520.

The conclusions of the original screen remain valid.

ENCLOSURE 6

Unit 2 – 10 CFR 50.59 Evaluation
Completed On: 07/31/2009

(Non-Proprietary)

EDISON

Operations Detail

SONGS

Order: NECP 800175663
Description: Steam Generator Replacement Mstr ECP U2

Operation No: 0520 **Sub Op:** **Op Wrkctr:** ED_SG SGRP
Description: 10CFR50.59 Evaluation
Work: 4.0 H **Number:** 1 **Duration:** 4.0 H
Risk level: **Potential Rx Impact:**
Risk Type:
RiskComments:

Operation Details:

10CFR50.59 Evaluation
NOTE: Completing a 50.59 Evaluation requires PQS T3EN99

Activity Title:

NECP 800071702, ASC D0018051 - Steam Generator Replacement - Unit 2

Activity Description:

The proposed activity replaces the design disclosure documentation and reference documentation for the SONGS Unit 2 Original Steam Generators (OSGs) with that for the Replacement Steam Generators (RSGs) and performs functional testing of the RSGs. This replacement and testing is required as a result of physically replacing the OSGs with RSGs. Having the OSGs replaced with the RSGs will improve the efficiency and reliability of Unit 2 by replacing a large number of plugged or otherwise degraded heat transfer tubes in each OSG with new tubes made from thermally-treated Alloy 690, which is less susceptible to degradation than the mill annealed Alloy 600 material used for OSG heat transfer tubing. Replacement of the steam generators is a replacement in-kind in terms of an overall fit, form, and function with no, or minimal, permanent modifications to the plant systems, structures, and components (SSCs).

The proposed activity covered by this evaluation includes the following:

1. Replacing the Original Steam Generator (OSG) documentation with the Replacement Steam Generator (RSG) documentation. Thich includes voiding/superseding the original Combustion Engineering/Westinghouse Electric Co. (CE/WEC) documents, as appropriate, and adding the new Mitsubishi Heavy Industries (MHI) design documents to the database.
2. Revising affected design bases documents (DBDs, UFSAR, calculations, etc.).
3. Revising other affected documents (90000 series documents, EQDPs, System Descriptions, Licensee Controlled Specifications (LCS), etc.).
4. Functional (performance) testing of the RSGs after installation.

Note 1: Activities associated with the physical removal of the OSGs and installation of the RSGs, such as creation and restoration of a temporary containment building construction opening, replacement of the steam generator insulation, re-calibration the steam generator level transmitters, or Cycle 16 Reload Analysis, are not

Operation Details Cont'd:

addressed herein, but instead are addressed in the specific NECPs governing these activities.

Note 2: This 10CFR50.59 Evaluation operation replaces operation 8000171702-50. The changes made in the responses reflect the additional scope of proposed changes to LCS 3.7.109, recognizing that periodic surveillance of compliance to the secondary side temperature limit is not required during the time period that the OSGs are drained for removal from the plant and RSGs are installed but have not been filled. The conclusion of the original 10CFR50.59 Evaluation remains valid.

A 10 CFR 50.59 Screen for the proposed activity (refer to 800175663-0510) determined that it involved changes to UFSAR-described methods of evaluation. These changes to methods of evaluation are summarized below.

1. The OSG stress analyses described in UFSAR Sections 3.9.1.2.2.1.11 and 3.9.1.2.2.2.3 utilized the ANSYS computer program, whereas the RSG analyses utilized the ABAQUS computer program.

2. The seismic analysis of the RVI for the RSGs was performed in accordance with the methodology described in Topical Report CENPD-178-P, Revision 1-P. The corresponding OSG analyses, described in UFSAR Section 3.7.3.14.1, reference Topical Report CENPD-178.

3. The RG 1.121 tube wall thinning evaluation for the RSGs changed the methods of evaluation from those described for the OSGs in UFSAR 5.4.2.3.1.3, as follows:

(a) The OSG analysis used the CEFLASH computer program for MSLB mass-energy blowdown analysis; the RSG analyses used manual calculation methods using the maximum differential pressure across the tube wall during the MSLB.
(b) The OSG LOCA analysis contained a two-step process utilizing the STRUDL and ANSYS computer programs to calculate displacement histories and tube stresses, respectively. The corresponding RSG analysis determined tube stresses from blowdown forces using only the ANSYS computer program.
(c) The OSG analyses considered primary loop branch line pipe break plus DBE [UFSAR 5.4.2.3.1.3.A] and Main Steam Line Break (MSLB) plus DBE [UFSAR 5.4.2.3.1.3.B] as separate events for determining tube stress. For the RSGs, the LOCA, DBE, and MSLB events were combined into one "limiting event" and the stresses for this combined event were calculated.

Summary of Evaluation:

A departure from a method of evaluation described in the UFSAR used in establishing the design bases or in the safety analyses, means changing any of the elements of the method described in the UFSAR, unless the results of the analysis are conservative or essentially the same.

RCS Structural Integrity Changing from ANSYS to ABAQUS

UFSAR Section 3.9.1.2.2.1.11 describes ANSYS as a large-scale, general-purpose, finite element program for linear and nonlinear structural and thermal analysis of reactor coolant loop components. The RCS structural integrity analyses for RSGs utilized ABAQUS in lieu of ANSYS. Like ANSYS, ABAQUS is a large-scale,

Operation Details Cont'd:

general-purpose finite element program for linear and nonlinear structural and thermal analysis. In using ABAQUS, elements of the approved OSG method of evaluation were changed, but the analyses did not use an entirely new method altogether.

ABAQUS and ANSYS were compared, using thermal and stress sample problems. The results of these sample analyses demonstrated that in all cases the ANSYS and ABAQUS results varied from theoretical solutions by no more than 1%, and ABAQUS and ANSYS results themselves were also within 1% of each other. A difference of 2% is considered within the margin of error for this type of analysis; hence ABAQUS produced results that are essentially the same as ANSYS, and does not constitute a departure from a method of evaluation described in the UFSAR used in establishing the design basis or in the safety analyses.

Seismic Analysis of Reactor Vessel Internals Changing from Topical Report CENPD-178 to CENPD-178-P, Revision 1-P

The original seismic analysis of the SONGS Unit 2 and 3 RVI (for the OSGs) was performed with the methodology described in C-E Topical Report CENPD-178 as referenced in UFSAR Section 3.7.3.14.1. Subsequent to submittal of CENPD-178, C-E modified modeling techniques, computer codes, testing methods, and acceptance criteria in response to changes in licensing requirements. As a result, the entire report was revised and resubmitted to the NRC as CENPD-178-P, Revision 1-P. CENPD-178-P, Revision 1-P, was approved by the NRC in NRC Letter, H. Bernard to A. Scherer, "Acceptance for Referencing of Topical Report CENPD-178 (P)," dated August 6, 1982.

Westinghouse, the current owner of the methodology, used it to perform the RSG RVI seismic analyses, consistent with its intended application, constraints and limitations. Consequently, its use does not constitute a departure from a method of evaluation described in the UFSAR used in establishing the design bases or in the safety analyses.

Tube Wall Thinning Analyses Changing from CEFLASH, STRUDL and ANSYS to Manual Calculations and ANSYS

The OSG analysis used the CEFLASH computer program for MSLB mass-energy blowdown analysis, whereas the RSG analysis used manual calculations to represent the MSLB blowdown loads by applying the maximum possible tube differential pressure. This manual calculation approach applies a differential pressure on the RSG tube wall that bounds the pressure calculated by CEFLASH.

Instead of using a two-step process for the OSGs LOCA analysis, utilizing computer programs STRUDL and ANSYS to calculate displacement histories and tube stresses, respectively, tube stresses for RSGs were determined from blowdown forces using only ANSYS. This use of ANSYS is consistent with its intended application, constraints and limitations as described in UFSAR 3.9.1.2.2.1.11, and, therefore, does not represent a departure from a method of evaluation.

For RSGs , the LOCA, DBE and SLB events were combined as one "limiting event" and the tube stresses were calculated for this event, whereas the OSG analyses considered PLPB plus DBE [UFSAR 5.4.2.3.1.3.A] and MSLB plus DBE [UFSAR 5.4.2.3.1.3.B] separately. Since the RSG approach combines the loads of three independent events (LOCA + DBE + SLB), rather than analyzing PLPB + DBE and MSLB + DBE loads

Operation Details Cont'd:

separately, the RSG method of evaluation is conservative relative to the OSG method of evaluation.

The results of the RSG tube wall thinning analysis are conservative or essentially the same as results from the UFSAR-described tube wall thinning analysis for the OSGs. In addition, this approach results in a lower tube wall plugging limit (44% for OSGs versus 35% for RSGs) which is also conservative. Therefore, the method of evaluation change for the tube wall thinning analysis is not a departure from a method of evaluation described in the UFSAR used in establishing the design bases or in the safety analyses.

Based upon the results of this evaluation:

__X__ Implement the activity per plant procedures without obtaining a License Amendment.

_____ Request and receive a License Amendment prior to implementation.

10CFR50.59 Evaluation

Completing a 50.59 Evaluation requires PQS T3EN99

SO123-XV-44.1 should be used to determine the content of each response (see Section 6.2 for additional guidance).

If the answer to any of the questions is "YES", then the proposed activity may not be implemented until a License Amendment has been obtained from the NRC.

UFSAR refers to the current FSAR, approved changes to the FSAR which have not yet been submitted to the NRC by amendment and documents included in the FSAR by reference.

1. Does the proposed activity result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the UFSAR/DSAR? (see 6.2.1)

N/A

2. Does the proposed activity result in more than a minimal increase in the likelihood of occurrence of a malfunction of an SSC important to safety previously evaluated in the UFSAR/DSAR? (see 6.2.2)

N/A

3. Does the proposed activity result in more than a minimal increase in the consequences of an accident previously evaluated in the UFSAR/DSAR? (see 6.2.3)

N/A

4. Does the proposed activity result in more than a minimal increase in the consequences of a malfunction of an SSC important to safety previously evaluated in the UFSAR/DSAR? (see 6.2.4)

Operation Details Cont'd:

N/A

5. Does the proposed activity create a possibility for an accident of a different type than any previously evaluated in the UFSAR/DSAR? (see 6.2.5)

N/A

6. Does the proposed activity create a possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated in UFSAR/DSAR? (see 6.2.6)

N/A

7. Does the proposed activity result in a design basis limit for a fission product barrier as described in the UFSAR/DSAR being exceeded or altered? (see 6.2.7)

N/A

8. Does the proposed activity result in a departure from a method of evaluation described in the UFSAR/DSAR used in establishing the design bases or in the safety analyses? (see 6.2.8)

No.

The SONGS 50.59 Resource Manual (Procedure SO123-XV-44.1), states that departure from a method of evaluation described in the UFSAR used in establishing the design bases or in the safety analyses, means changing any of the elements of the method described in the UFSAR, unless the results of the analysis are conservative or essentially the same. One example of a change that is not considered a departure from a method of evaluation described in the UFSAR is:

"Use of a methodology revision that is documented as providing results that are essentially the same as, or more conservative than, either the previous revision of the same methodology or another methodology previously accepted by NRC through issuance of an SER."

The 50.59 Resource Manual continues:

"Analytical results obtained by changing any element of a method are conservative relative to the previous results, if they are closer to design bases limits or safety analyses limits (e.g., applicable acceptance guidelines).

"For example, a change from 45 psig to 48 psig in the result of a containment peak pressure analysis (with design basis limit of 50 psig) using a revised method of evaluation would be considered a conservative change when applying this criterion. In other words, the revised method is more conservative if it predicts more severe conditions given the same set of inputs. This is because results closer to limiting values are considered conservative in the sense that the new analysis result provides less margin to applicable limits for making potential physical or procedure changes without a license amendment."

Operation Details Cont'd:

RCS Structural Integrity Changing from ANSYS to ABAQUS

ANSYS was the computer program used to perform the OSG RCS structural integrity analyses. UFSAR Section 3.9.1.2.2.1.11 describes ANSYS as a large-scale, general-purpose, finite element program for linear and nonlinear structural and thermal analysis of reactor coolant loop components.

In evaluating the impact of the RSGs, the RCS structural integrity analyses utilized the ABAQUS computer program. Like ANSYS, ABAQUS is a large-scale, general-purpose finite element program for linear and nonlinear structural and thermal analysis. In using ABAQUS in lieu of ANSYS for these analyses, elements of the approved OSG method of evaluation were changed, but the analyses did not use an entirely new method altogether.

Only linear elastic elements of ABAQUS for were utilized the RSG analyses. One analysis, for the tube-to-tubesheet seal weld, L5-04GA431 (SO23-617-1-C1306), includes an ABAQUS model that contains gap elements to account for boundary conditions above the seal weld. When gap elements are used in a model combined with time history analysis of dynamic loading, they can have an energy dissipating effect that changes the results (if the gaps open and close in the response time history). If loads are applied as dynamic loads, these would represent nonlinear aspects of the analysis.

In the tube-to-tubesheet seal weld analysis, the discussion and tables of transient loads, which are primarily thermal loads, should not be taken to imply that any loads were applied as dynamic loads. These tables were used only to find the maximum temperature difference to apply statically to the model.

Essentially, the gap elements were used as a convenience that replaced having to manually iterate the model to perform a displacement compatibility check (i.e., confirm that displaced elements do not physically overlap). This practice is well-known and accepted in the finite element analytical arena, and is still well within the linear elastic range of modeling. In essence, one can get the same exact results by using a computer program without gap element capability (e.g., a simple linear elastic computer program), and manually iterating node connectivity so that the displacements are compatible.

Additionally, the analyses of the tube bundle assembly did not model spaces between tubes and anti-vibration bars as gap elements. These spaces were either too small or were conservatively considered to be too large to interact with a tube (thereby providing no tube support). As such, the RSG analyses used only linear elastic analysis capabilities of ABAQUS.

To demonstrate that this element change does not represent a departure from a method of evaluation, one must compare the results of sample runs made using each program and demonstrate that the new program yields results that are conservative or essentially the same as those obtained using the UFSAR-described program. ABAQUS and ANSYS were compared, using sample thermal analysis and stress analysis problems. The analyzed structures were a thick-walled cylinder, a thick-walled sphere, and a curved beam of varying dimensions. The cylinder model included 2-D axisymmetric, 3-D shell, and 3-D solid elements. The sphere model included 2-D axisymmetric and 3-D solid elements. The thermal analysis problem included steady-state and non steady-state loads. The stress analysis problem included internal pressure, thermal, and concentrated loads. Loading conditions varied, but were applied to the corresponding structural model consistently.

Operation Details Cont'd:

Theoretical solutions were also derived for comparison purposes.

The results of these sample analyses (contained in MHI document number L5-04GA441 [SO23-617-1-M1382]) were compared to each other, as well as to the theoretical solutions. In all cases, the ANSYS and ABAQUS results varied from the theoretical solutions by less than 1%, and ABAQUS and ANSYS results were also within 1% of each other. A difference of 2% is considered within the margin of error for this type of analysis; hence, ABAQUS produces results that are essentially the same as those produced by ANSYS. Therefore, the use of the ABAQUS computer program for the RCS structural analysis does not constitute a departure from a method of evaluation described in the UFSAR used in establishing the design basis or in the safety analyses.

Seismic Analysis of Reactor Vessel Internals Changing from Topical Report CENPD-178 to CENPD-178-P, Revision 1-P

The original seismic analysis of the SONGS Unit 2 and 3 RVI was performed in accordance with the methodology described in Topical Report CENPD-178, as referenced in UFSAR Section 3.7.3.14.1. CENPD-178 was originally submitted to the NRC in 1976 as a Combustion-Engineering Topical Report. Subsequent to submittal of CENPD-178, C-E modified modeling techniques, computer codes, testing methods, and acceptance criteria in response to changes in licensing requirements. As a result, the entire report was revised and resubmitted to the NRC as CENPD-178-P, Revision 1-P. CENPD-178-P, Revision 1-P, was approved by the NRC in NRC Letter from H. Bernard to A. Scherer, "Acceptance for Referencing of Topical Report CENPD-178 (P)," dated August 6, 1982.
Westinghouse, the current owner of the methodology, used it to perform the RVI seismic analyses for RSGs, consistent with its intended application, and constraints and limitations. Consequently, its use does not constitute a departure from a method of evaluation described in the UFSAR used in establishing the design bases or in the safety analyses.

Tube Wall Thinning Analyses Changing from CEFLASH, STRUDL and ANSYS to Manual Calculations and ANSYS

The R.G. 1.121 tube wall thinning analysis methodologies applied to the OSGs are described in UFSAR Section 5.4.2.3.1.3. The corresponding analysis for the RSGs involved changes to methods of evaluation.

OSG Analysis

The UFSAR-described methodology considered two separate loading conditions, denoted "PLPB + DBE" and "MSLB + DBE," which consisted of two separate combinations of pressure, PLPB (Primary Loop Pipe Break or LOCA), DBE (Design Basis Earthquake), and MSLB (Main Steam Line Break) loadings.

UFSAR Section 5.4.2.3.1.3.A describes the PLPB + DBE analysis performed for the OSGs. The CEFLASH computer program was used to perform the OSG hydraulic dynamic analysis of the primary loop during the PLPB event. CEFLASH produced load time histories on the tube bundle. From these load time histories, the computer program STRUDL was used to calculate tube displacement time histories. Next, peak tube stresses from the PLPB loads were calculated, using the computer program ANSYS, and then combined with DBE tube stresses using the square-root-of-the-sum-of-the-squares (SRSS) method. Finally, tube stresses due to pressure load were added to the combined PLPB and DBE stresses.

Operation Details Cont'd:

The same method was used for the MSLB + DBE analysis, as described in UFSAR Section 5.4.2.3.1.3.B. CEFLASH was used for the hydraulic dynamic analysis of the secondary loop during a postulated MSLB accident. Tube displacement time histories at various elevations in the SG were calculated using STRUDL. ANSYS was then used to calculate tube stresses. DBE stresses were calculated and combined with MSLB stresses by SRSS, then added to pressure stresses.

The results of the PLPB + DBE and the MSLB + DBE analyses were compared and tube stresses for the governing condition (PLPB + DBE) were compared to the ASME Code allowables .

RSG Analysis

The R.G. 1.121 tube wall thinning analysis for the RSGs considered a single limiting combination of design basis events: LOCA, DBE and MSLB. CEFLASH was originally used to perform the hydraulic dynamic analysis of the primary loop during a LOCA event to determine dynamic structural response of the steam generators (OSGs) to the impulsive loading imparted by the escaping fluid.

For the RSGs, a structural model of the tube bundle was generated using the ANSYS computer code in the seismic and LOCA rarefaction analyses. New seismic response spectra were developed for the RSGs. The rarefaction wave analysis utilized the OSG pressure-time history (dynamic loadings) developed for the OSGs using CEFLASH. For stress calculations, the maximum seismic and LOCA rarefaction wave loads were combined using the SRSS method.

The MSLB event was conservatively modeled assuming the primary side at the design pressure (the pressurizer safety valve set point), plus 3% accumulation, and the secondary side at atmospheric pressure. The stresses due to MSLB differential pressure in the tubes were added directly to the combined stresses calculated for the seismic and LOCA loads. Thus, the RSG analysis considered LOCA loads, DBE loads, and maximum MSLB primary-to-secondary differential pressure across the tubes simultaneously, which is conservative.

Comparison of OSG to RSG Methods

The OSG and RSG analyses used the same computer program, CEFLASH, for hydraulic dynamic analysis of the primary loop during the primary loop pipe break event..

Instead of using the UFSAR-described two-step process with the STRUDL and ANSYS computer programs to calculate displacement histories and tube stresses, respectively, the RSG analysis determined tube stresses from blowdown forces using only ANSYS. ANSYS is described in UFSAR Section 3.9.1.2.2.1.11 as a large-scale, general-purpose, finite element program for linear and nonlinear structural and thermal analysis of reactor coolant loop components. This use of ANSYS for RSGs is consistent with its intended application, and constraints and limitations as described in UFSAR 3.9.1.2.2.1.11. Therefore, utilizing only ANSYS instead of STRUDL and ANSYS is not a departure from the UFSAR-described method of evaluation.

In contrast to the OSG MSLB + DBE analysis, which used the computer program CEFLASH for the mass-energy blowdown analysis portion, the RSG analysis did not include dynamic hydraulic analysis of the MSLB. Instead, the peak secondary side blowdown forces were conservatively calculated by manually adding the loads resulting

Operation Details Cont'd:

from the maximum primary to secondary differential pressure across the tube walls to the LOCA + DBE loads. This manual calculation approach applies a differential pressure on the tube wall that bounds the differential tube wall pressure calculated by CEFLASH.

Finally, for the RSGs, the tube stresses due to LOCA, DBE and MSLB were combined as one "limiting event," whereas the OSG analyses considered PLPB + DBE and MSLB + DBE separately. By inspection, the RSG approach is conservative relative to the OSG method of evaluation.

This conservatism is further apparent by comparing the allowable tube wall plugging criteria contained in Technical Specification 5.5.2.11 for the OSGs (44%) to the criteria proposed in PCN-583 for the RSGs (35%). Clearly, the analysis method used for the RSGs results in a lower plugging limit, which is conservative.

Based on the above, it can be concluded that the results of each element of the RSG tube wall thinning analysis are conservative or essentially the same as results from the UFSAR-described tube wall thinning analysis for the OSGs. Therefore, the RSG tube wall thinning analysis is not a departure from a method of evaluation described in the UFSAR used in establishing the design bases or in the safety analyses.

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(NSG Review Comments)
Document below any issues wherein the NSG reviewer either disagrees with the response to a question, or where there is inadequate detail provided to support the conclusion:

Does the change require prior NRC approval?

ENCLOSURE 7

Unit 3 – 10 CFR 50.59 Screening
Completed On: 09/18/2009

(Non-Proprietary)

EDISON

Operations Detail

SONGS

Order: NECP 800175664
Description: Steam Generator Replacement Mstr ECP U3

Operation No: 0190 **Sub Op:** **Op Wrkctr:** ED_SGD SGRP Design
Description: 10CFR50.59 Screening
Work: 4.0 H **Number:** 1 **Duration:** 4.0 H
Risk level: **Potential Rx Impact:**
Risk Type:
RiskComments:

Operation Details:

10CFR50.59 Screening
(Previous Operation 800175664-0030)

NOTE: *Completing a 50.59 Screen requires PQS T3EN98*

SECTION I: GENERAL INFORMATION (See Keypoints Section I)

1. Primary Document Number and Title:

NECP 800071703, Steam Generator Replacement # Unit 3

2. Description of Proposed Activity:

The proposed activity replaces the design disclosure documentation and reference documentation for the SONGS Unit 3 Original Steam Generators (OSGs) with that for the Replacement Steam Generators (RSGs) and performs functional testing of the RSGs. This replacement and testing is required as a result of physically replacing the OSGs with the RSGs Having the OSGs replaced with the RSGs will improve efficiency and reliability of Unit 3 by replacing a large number of plugged or otherwise degraded heat transfer tubes in each OSG with new tubes made from thermally-treated Alloy 690, which is less susceptible to degradation than the mill-annealed Alloy 600 material used for OSG heat transfer tubing. Replacement of the steam generators is a replacement in-kind in terms of an overall fit, form and function with no, or minimal, permanent modifications to the plant systems, structures or components (SSCs).

The proposed activity covered by this screen includes the following:

·Replacing the Original Steam Generator (OSG) documentation with the Replacement Steam Generator (RSG) documentation, which includes voiding/superseding the original Combustion Engineering/Westinghouse Electric Co. (CE/WEC) documents, as appropriate, and adding the new Mitsubishi Heavy Industries (MHI) design documents to the database.
·Revising affected design bases documents (DBDs, UFSAR, calculations, etc.).
·Revising other affected documents (90000 series documents, EQDPs, System Descriptions, Licensee Controlled Specifications (LCS), etc.).
Functional (performance) testing of the RSGs after installation.

Operation Details Cont'd:

NOTE:
Activities associated with the physical removal of the OSGs and installation of the RSGs, such as creation and restoration of a containment building temporary construction opening, replacement of the steam generator insulation, re-calibration of the steam generator level transmitters or Cycle 16 Reload Analysis, are not addressed herein, but instead are addressed in the specific NECPs governing these activities.

The RSG design bases are consistent with those of the OSGs, except for the specific values of the major parameters, and are as follows (see UFSAR Sections 4.4, 5.1, 5.2, 5.4 and 10.1):

·The RSGs are designed and fabricated as ASME B&PV Code Section III, Class 1 components, based on the 1998 Edition through 2000 Addenda. The OSGs are designed and fabricated as Class A components in conformance with the ASME B&PV Code, Section III, 1971 Edition plus Addenda through summer 1971.
·Each RSG is designed to remove the rated heat load of 1729 MWt from the reactor coolant system (RCS), which is the same as the OSG current operating heat load.
·Each RSG is designed to remove the rated heat load with up to 8% (779) tubes plugged, which is the same as the design plugging limit for the OSGs.
·Each RSG is designed to produce 7.588E6 lb/hr (vs. 7.414E6 lb/hr for OSGs) of 833 psia (vs. 900 psia for OSGs) saturated steam with less than 0.1% (vs. 0.2% for OSGs) moisture content when supplied with feedwater at 442oF.

NOTES:
The saturated steam pressure of 833 psia is the best estimate pressure at the steam generator outlet nozzle with the reactor coolant inlet (Thot) temperature of 598 oF (corresponding to the Tcold of 541.3oF), reactor coolant best estimate flow rate (79.79E6 lb/hr), no (0%) tubes plugged and an assumed fouling factor ([ ] ft2hroF/Btu). The actual pressure will vary depending on the actual values of these parameters during plant operation.

The RSGs are qualified to operate in the Thot range from 598 to 611oF, which corresponds to the Tcold range from 541.3 to 555.4oF. However, NECP 800071703 and this 50.59 screen only allows the plant to operate up to Thot <= 598oF, as additional analyses are required to be performed to evaluate the rest of the RCS and support systems' ability to operate above Thot > 598oF.

The major physical differences between the RSGs and OSGs are as follows:

·The RSGs have a greater number of tubes (9,727 vs. 9,350) and a larger heat transfer surface area than the OSGs ([ ] ft2 vs. ~ [ ] ft2).
·The RSG reactor coolant volume is greater than the OSG volume (2003 ft3 vs. 1895 ft3).
·The RSG tube wall thickness is less than the wall thickness of the OSG tubes (0.0429 in. vs. 0.048 in.).

·The RSG tubes are Alloy 690 (thermally-treated) while the OSG tubes are Alloy 600 (mill-annealed).
·The RSG feedwater ring is fabricated from erosion-corrosion resistant Cr-Mo alloy steel with Alloy 690 TT fittings, whereas the OSG feedwater ring is made of carbon steel (with the exception of the flow distribution box).

Order: NECP **800175664**
Description: **Steam Generator Replacement Mstr ECP U3**
Operation No: 0190 **Sub Op:** **Op Wrkctr:** ED_SGD SGRP Design

Operation Details Cont'd:

·All RSG tubes are U-bend shape, whereas the OSG tubes have both U-bend shape (inner rows of the tube bundle) and square-bend shape (outer rows of the tube bundle).
·The RSG channel head has a flat bottom, thicker divider plate, as compared to the OSGs, and no stay cylinder.
·The RSGs have integral grooves on all primary nozzles for nozzle dam mounting, whereas the OSGs do not.
·The RSG tube-to-tubesheet joints are seal-welded and hydraulically expanded with a one-step mechanical roll at the primary face, while the OSG joints are seal-welded and explosively expanded.
·The RSG tube supports consist of 7 broached tube support plates in the straight-leg region and anti-vibration bars in the U-bend region, while the OSG tube supports consist of the egg-crate type supports in the straight-leg region and batwings and vertical strips in the U-bend region.
·The RSG feedwater ring has a #gooseneck# preventing watethammer and employs perforated spray nozzles mounted on top of the ring, whereas the OSGs do not have the #gooseneck and employj-nozzles.
·The RSGs have [ ] moisture separators and [] banks of steam dryers each, whereas the OSGs have [ ] moisture separators and [ ] steam dryers.
·The RSG steam nozzle is an integrally forged part of the upper head (i.e. there is no weld seam between the nozzle and the head) and has 7 integral venturi nozzles designed to limit the steam flow and reduce the rate of energy release into the containment during a postulated main steam line break (MSLB) inside containment. The OSGs have no such flow limiting device.
·The number and orientation of the RSG major nozzles and instrument taps are the same as for the OSGs, except that: the RSGs have one wide range level lower tap on the extension ring, whereas the OSGs have such a tap installed on the handhole cover; the circumferential orientation of two narrow range level taps on the RSGs is different from that on the OSGs to avoid interference between the taps and one of the lifting trunnions; the RSGs have one each secondary side sampling tap, dry lay-up nozzle, wet lay-up nozzle, and recirculation nozzle on the upper shell, whereas the OSGs have only the sampling tap; the RSGs have one blowdown/secondary drain nozzle located on the tubesheet, whereas the OSGs have such a nozzle on the lower shell above the tubesheet.
·The RSGs have a greater number of secondary side handholes and inspection ports [] at [] in. versus 2 at 6 in. and [ ] at [ ] in. versus none, respectively).
·The RSGs have two lifting trunnions on their upper shell, separate from the secondary manways, for installation rigging. The OSGs had no separate lifting provision and utilized the secondary manways for installation rigging.
·The RSG lower shell outside diameter is 174.6 in., whereas the OSG diameter is [ ] in.
·The RSGs have a larger primary manway inside diameter than the OSGs ([] in. vs. 16 in.).
·RSG drv. flooded and operating weights are greater than the correspondina weights of the OSGs ([ ] lb. vs. [ ] lb., [ ] lb. vs. [ ] lb. and 1,548,700 lb. vs. [ ] lb, respectively).
·To maintain the SG within its fracture toughness limits, when the RSG secondary pressure is greater than 200 psig, the RSG steam generator secondary coolant (main or auxiliary feedwater) temperature must be less than [] oF, whereas the Unit 3 OSGs needed a more restrictive [ ] oF) limit (This scope applies to Unit 3 SGs only). [The Unit 3 OSG limitation of [ ] oF is based on a Nil Ductility Reference Temperature (RTndt) of [ ] oF, whereas the RSG limitation of [] oF is based on an RTndt of [] oF (ref. Unit 3 LCS 3.7.109).]
·A change was also made to LCS 3.7.109 to recognize that periodic surveillance of compliance to the secondary side temperature limit is not required during the time that the OSGs are drained for removal from the plant (with the core off-loaded) and the time the RSGs have been installed and may be pressurized (prior to reloading fuel in the core).

Operation Details Cont'd:

The proposed activity also involves updates of certain UFSAR-described analyses that demonstrate that with the RSGs installed the consequences of accidents would not result in exceeding the dose limits, or demonstrate that the intended design functions will be accomplished. In addition to the discussion of the RSG component, Sections I.3, II.1, II.3 and II.5 of this 50.59 Screen include the following subsections to address the effects of the proposed activity on the design functions, performance aspects, methods of evaluation, and technical specifications/LCS of/for the plant systems, structures or components affected by the proposed activity:

Subsection (a) - RCS Structural Integrity
Subsection (b) - ECCS Performance
Subsection (c) - Non-LOCA Transients
Subsection (d) - Containment P-T Analysis
Subsection (e) - Low Temperature Overpressure Protection
Subsection (f) - RPS, ESFAS, COLSS and CPC
Subsection (g) - NSSS Performance
Subsection (h) - Non-Safety Related Control Systems Performance

3. Design Function(s) and/or Method(s) of Evaluation:

The design functions of the steam generators are to:

1. Function as a part of the reactor coolant pressure boundary (RCPB).
2. Transfer heat between the RCS and main steam system.
3. Remove heat from the RCS to achieve and maintain safe shutdown following design basis accidents (except for a large break LOCA) and other UFSAR-described events.

The design functions of the steam generator tubes and tube supports are to:

1. Limit tube flow-induced vibration and reactor coolant pump-induced vibration to acceptable levels during normal operating conditions.
2. Withstand blowdown forces from severance of a steam nozzle and ensure that ASME Code allowable stress limits are met.
3. Maintain acceptable ASME Code stress levels under design basis accident conditions (i.e., to prevent a tube rupture concurrent with other accidents, and to prevent multiple tube ruptures during a postulated single steam generator tube rupture event), and
4. Function as a part of the RCPB.

The UFSAR-described methods of evaluation for the steam generators are:

1. UFSAR Sections 3.9.1.2.2.1.11 and 3.9.1.2.2.2.3 state that the stress analysis and dynamic analysis of RCS components are performed by finite element method using the ANSYS computer program. Specifically, the finite element capabilities of the program are used to determine the primary plus secondary stresses and peak stresses due to normal and upset conditions and to evaluate the resulting stress intensities.

Operation Details Cont'd:

2. UFSAR 5.4.2.3.1.3 describes specific methods of evaluation for the Regulatory Guide (RG) 1.121 tube-wall thinning evaluation, using the following computer programs:
CEFLASH for blowdown analysis
STRUDL for tube displacement response analysis, and
ANSYS for tube stress analysis.

Subsection (a) - RCS Structural Integrity

RCS design functions considered in this subsection are:
(1)Maintain structural integrity of the main coolant loop (MCL) piping (including tributary line nozzles and partial penetration nozzles), surge line (including partial penetration nozzles), reactor vessel (RV) and its supports, reactor coolant pumps (RCPs) and their supports, steam generator supports, reactor vessel internals (RVI), pressurizer, control element drive mechanisms (CEDMs), head lift rig (HLR), in core instrumentation (ICI) tubes, Reactor Vessel Gas Vent System (RVGVS), and fuel and control element assemblies (CEAs).
(2)Maintain the capability to insert the control element assemblies (CEAs).

The related UFSAR-described methods of evaluation are:

(1)UFSAR Section 3.9.1.2.2.1.11 describes ANSYS as a large-scale, general-purpose, finite element program for linear and nonlinear structural and thermal analysis of reactor coolant loop components.
(2)Seismic analysis of the reactor internals is discussed in UFSAR Section 3.7.3.14 and 3.7.3.14.1, which references Combustion Engineering Topical Report CENPD-178.

Subsection (b) - ECCS Performance

The design functions of the ECCS are to mitigate the effects of a large break loss of coolant accident (LBLOCA) and a small break loss of coolant accident (SBLOCA) and to maintain post-LOCA long term cooling to meet the acceptance criteria of 10 CFR 50.46 [UFSAR 6.3.3].

The methods of evaluation for ECCS performance analyses are explicitly described in UFSAR 6.3.3.2 (LBLOCA effects), 6.3.3.3 (SBLOCA effects), and 6.3.3.4 (post-LOCA long term cooling), which includes a citation to NRC-approved computer programs and analytical methods documented in Combustion Engineering Topical Report CENPD-254-P-A.

Subsection (c) - Non-LOCA Transients

The design functions of the steam generators and the systems, structures and components (SSCs) credited for mitigating the effects of the non-LOCA transients, per UFSAR Sections 5.4.2.1 and 5.2, are:
(1)Transfer heat from the primary to the secondary system as required for plant modes in which the steam generators must be operable, and
(2)Provide a part of the reactor coolant pressure boundary.

The UFSAR-described method of evaluation for non-LOCA transient events utilizes the CESEC computer program for loss of all normal feedwater flow (UFSAR 15.10.2.2.5) and loss of all feedwater flow with a concurrent single failure of an active component (UFSAR 15.10.2.3.2).

Operation Details Cont'd:

Subsection (d) - Containment P-T Analysis

The design functions relevant to this subsection are:
(1)UFSAR Table 6.2-3 states that the containment internal design pressure is 60.0 psig and the containment design temperature is 300°F. Technical Specification (TS) Bases B3.6.5 clarifies that the temperature limit of 300°F is not an ambient atmosphere temperature limit, but rather pertains to the containment structure such as the containment liner plate and concrete. Per UFSAR Section 6.2.2.1.3, the design temperature of equipment subjected to LOCA or MSLB conditions is 300°F. Per UFSAR Table 6.2-34 the containment emergency cooling unit (ECU) atmosphere inlet temperature is limited to 300°F.
(2)UFSAR Sections 6.2.1.1.1.4, 6.2.2.1.1C and 6.2.2.2.1A, state that the containment emergency fan cooler system, in conjunction with the Containment Spray System (CSS) and the shutdown heat exchangers (i.e., one train of each system), is capable of reducing the post-LOCA containment pressure from the peak value to one-half of the peak value in 24 hours in accordance with 10 CFR 50, Appendix A, GDC 38 and Standard Review Plan 6.2.1.1.A.

The UFSAR-described methods of evaluation are:
(1)UFSAR Section 6.2.1.1.3.1.C states that the containment pressure analyses are performed using the COPATTA computer program and specifically describes some aspects of the COPATTA modeling.
(2)UFSAR Section 6.2.1.3.3 states that the LOCA blowdown phase is simulated with the CEFLASH 4A computer program and specifically describes some aspects of the CEFLASH 4A modeling.
(3)UFSAR Section 6.2.1.3.4 states that the LOCA core reflood and post-reflood phase is simulated with the FLOOD3 computer program and specifically describes some aspects of the FLOOD3 modeling.
(4)UFSAR Section 6.2.1.3.5 states that the LOCA long term phase M-E release is analyzed with the use of the COPATTA and CONTRANS computer programs and specifically describes some aspects of the decay heat model.
(5)UFSAR Section 6.2.1.4 states that the MSLB M-E data is analyzed with the use of the SGNIII computer program and UFSAR Section 6.2.1.4.4 specifically describes some aspects of the SGNIII modeling.
(6)UFSAR Section 6.2.1.4.3 states that a 2 percent increase in the initial inventory resulting from thermal and pressure expansion of the steam generator at operating conditions is to be included in the analysis.
(7)Various locations within UFSAR Section 6.2 include references to the M-E and peak P-T analysis power level being 3390 MWt plus 2 percent uncertainty (although the plant now operates at 3458 MWt).

Subsection (e) - Low Temperature Overpressure Protection (LTOP)

The design function relevant to this subsection is the ability of the shutdown cooling system (SDCS) relief valve to terminate any inadvertent pressure transient occurring when the RCS temperatures are below the applicable P-T operating curve temperatures, corresponding to pressures below the relief valve setpoint pressure, as described in UFSAR, Section 5.2.2.11.2.2. Above the LTOP enable temperature, overpressure protection is provided by the pressurizer safety valves (as the SDCS relief valve is isolated from the RCS).

The UFSAR describes mass and energy addition transients analyzed to demonstrate acceptable LTOP for the existing plant configuration. The methods used to analyze the LTOP transients are not described in the UFSAR.

Subsection (f) - RPS, ESFAS, COLSS and CPC

Operation Details Cont'd:

The design functions pertinent to this subsection are:
(1)The Reactor Protective System (RPS) trips the reactor to protect the core fuel design limits and reactor coolant system pressure boundary for anticipated operational occurrences, and also assist in limiting conditions for certain accidents (UFSAR Section 7.2.1.1).
(2)The Engineered Safety Features Actuation System (ESFAS) generates signals that actuate the Engineered Safety Features (ESF) systems (UFSAR Section 7.3).
(3)The Core Operating Limit Supervisory System (COLSS) monitors the limiting conditions for operation on peak linear heat rate, margin to departure from nucleate boiling, total core power, axial shape index and azimuthal tilt (UFSAR Section 7.7.1.5.1).
(4)The Core Protection Calculators (CPCs) calculate departure from nucleate boiling ratio and linear power density and initiate a trip on those parameters (UFSAR Section 7.2.1.1.2.5).

UFSAR Section 7.2.1.2, "Design Bases," states: "A detailed description of the methodology used to calculate the setpoints, including their explicit uncertainties, is provided in the report CEN 112(S) Revision 0, Plant Protection System Selection of Trip Setpoint Values, November 15, 1979."

Subsection (g) - NSSS Performance

The design functions of the affected systems are:
(1)The RCS circulates water in a closed cycle, removing heat from the reactor core and internals and transferring it to the secondary (steam generating) system (UFSAR Section 5.1).
(2)The CVCS maintains the chemistry, purity and activity level of the reactor coolant within limits, maintains the required volume of water in the RCS, provides a controlled discharge path for reactor coolant to the radwaste system, controls RCS boron concentration, provides auxiliary pressurizer spray, provides means for testing the check valves that isolate the safety injection system from the RCS, provides continuous measurement of reactor coolant boron concentration, collects reactor coolant pump seal controlled bleed-off, provides for leak testing the RCS, provides a normal and an alternate means for filling the RCS (UFSAR Section 9.3.4.1.1).
(3)In the event of a LOCA, the SIS provides core cooling sufficient to prevent significant alteration of core geometry, precludes fuel melting, limits the cladding metal water reaction, removes energy generated in the core for an extended period of time and maintains the core sub-critical for the duration of the LOCA (UFSAR Section 6.3.1).
(4)The SDCS removes heat from the RCS during post-shutdown periods (UFSAR Section 5.4.7.1).
(5)The CSS removes heat from the containment atmosphere in the event of a LOCA or MSLB inside containment (UFSAR Section 6.2.2.1).

The methods of evaluation for NSSS performance analyses are not described in the UFSAR.

Subsection (h) - on-Safety Related Control Systems Performance

The design functions of the affected systems are:
(1)The Pressurizer Pressure Control System (PPCS) maintains pressurizer pressure within specified limits (UFSAR Section 7.7.1.2.1).
(2)The Pressurizer Level Control System (PLCS) regulates Reactor Coolant System (RCS) water inventory (UFSAR Section 7.7.1.2.2).

Operation Details Cont'd:

(3)The Feedwater Control System (FWCS) maintains steam generator water level within acceptable limits (UFSAR Section 7.7.1.3).
(4)The Steam Bypass Control System (SBCS) provides a means for controlling NSSS thermal energy during plant startup, cooldown and hot standby, and accommodates load rejections, unit trips and other conditions that result in generation of excess energy by the NSSS (UFSAR Section 7.7.1.4.1).

The Methods of Evaluation for the non-safety control system performance are not described in the UFSAR.

SECTION II: 50.59 SCREEN CONCERNS (See Keypoints Section II)

State if the proposed activity:

1. Changes an SSC in a manner that adversely affects the UFSAR/DSAR design function(s) or has an adverse affect on the method of performing or controlling UFSAR/DSAR design function(s).

No.
Replacement of the OSGs with the RSGs is a replacement in-kind in terms of form, fit, and function, and is designed to involve no, or only minimal, permanent modifications to the plant structures, systems or components. In fact, installation of the RSG requires only minor modifications to the blowdown piping routing, instrument sensing line routing, access platform configuration and possibly steam generator key bracket configuration.

The RSGs are designed and fabricated as ASME Section III, Class 1 components in accordance with the requirements of the newer ASME Code edition than that used to design and fabricate the OSGs. During fabrication, the steam generator parts and assemblies were non-destructively examined and tested in accordance with the ASME Code, Section III requirements. The steam generators were also subject to additional inspections and tests beyond the Section III requirements. These included ultrasonic testing of tubesheet cladding for defects, ultrasonic testing of weld cladding for bond integrity, and helium leak rate testing of the tube-to-tubesheet welds. Shop hydrostatic tests of the primary and secondary side of the steam generators were conducted in accordance with the ASME Code, Section III, and as replacement components the RSG were subject to pre-service examinations in accordance with ASME Code, Section XI requirements. The fact that the RSG were designed, fabricated and examined to the newer edition of the Code is an enhancement over the OSGs.

The RSGs are designed to perform the same design functions as those currently performed by the OSGs (see Section I.3), and the differences listed in Section I.2 do not affect their ability to perform these design functions. Specifically, installation of the RSGs does not require any changes to the steam generator water level, Feedwater Control System (FWCS), ADV or MSSV setpoints.

The values of the RSG major design parameters are different than the values of the corresponding OSG parameters (see Section I.2). The RSG steam flow is slightly higher, the outlet steam pressure is lower and the moisture content is considerably lower than the values for the OSGs. These changes are in a conservative direction (pressure) or constitute an improvement over the OSGs (moisture content).

The RSG design parameters are achieved at the same feedwater and reactor coolant inlet (Thot) temperatures

Operation Details Cont'd:

as the current OSG operating temperatures, with an approximately the same NSSS reactor coolant flow rate as the rate currently measured with the OSGs installed. However, it is achieved with a considerably larger heat transfer surface area.

The RSG heat transfer area is larger than the OSG area ([ ] ft2 vs. ~ [ ] ft2) and the RSG tube bundle is taller than the OSG bundle. The effects of the larger tube bundle are documented in Parts 3 and 4 of Reference 1. The larger and taller RSG tube bundle offers an increased driving head for natural circulation flow, so that the RSGs will perform the same as, or better than, the OSGs during the events that credit natural circulation. The steam flow limiter integral to the RSG steam outlet nozzle does not impact natural circulation cooldown, as the Atmospheric Dump Valve (ADV) discharge port size is limiting for RCS cooldown.

The RSG primary side volume is larger than the OSG volume (2003 ft3 vs. 1895 ft3). Due to this increase, the time necessary to cooldown and depressurize the RCS to Shutdown Cooling System (SDCS) entry conditions (approximately 350oF and 376 psia) will be extended slightly. This time increase is insignificant when compared to the overall cooldown time and the available Technical Specification inventory in the Condensate Storage Tanks.

Likewise, the RCS volume increase results in an insignificant time extension for the SDCS to reduce RCS temperature. Specifically, it will take an additional 1.2 minutes to reach a refueling temperature of 130oF for dual-train SDCS operation (4.2 minutes for single-train operation). Cold shutdown is defined as the RCS temperature being 200oF, which takes much less time than cooling to 130oF. This slight time increase does not impact compliance with Branch Technical Position (BTP) Reactor Systems Branch (RSB) 5-1 (i.e., achieving cold shutdown conditions within 24 hours from inside the control room with a concurrent single failure). Additionally, the time increase does not adversely affect the ability to achieve and maintain post-fire safe shutdown (cold shutdown) within 72 hours following the fire (10CFR50, Appendix R, III.L), because this time increase is negligible compared to the shutdown time frame in the RSB 5-1 and Appendix R events.

The RCS volume increase will also result in a slight increase of the containment flooding level, following a LOCA. This increase was evaluated from the perspective of an impact on the instruments required for post-LOCA response and it was concluded that the change in the flooding level was acceptable (Reference Calc. N-4060-030, ECN A54375).

The RSG tube wall is thinner than the OSG tube wall (0.0429 in. vs. 0.048 in.). Regulatory Guide 1.121 analysis of the RSG tubes was performed consistent with the loading conditions and acceptance criteria guidance provided in the guide and includes assessment of the loads that result from postulated accident conditions, including a design basis earthquake (DBE) in combination with a LOCA and MSLB. The analysis concluded that a tube would have to be plugged if it contained a flaw to a depth lesser than that for the OSGs (35% vs. 44%). This reduction of the tube plugging limit is conservative.

The RSG major materials of construction are different than the OSG materials. These differences represent a vast improvement over the OSG materials in terms of corrosion, erosion-corrosion and wear resistance. The RSG primary and secondary side pressure boundary materials have nil-ductilitv (ndt) reference temperature (RT) the same as or better than those for the Unit 3 OSG materials ([] oF, versus a [ ] oF limit in Unit 3 LCS 3.7.109). The Unit 3 OSG limitation of [ ] oF is based on an RTndt of [ ] oF, whereas the RSG limitation of [] oF is based on an RTndt of [] oF (ref. Unit 3 LCS 3.7.109). Lowering the RTndt reflects a corresponding improvement in

Operation Details Cont'd:

material properties which is not adverse.

A change was also made to LCS 3.7.109 to recognize that periodic surveillance of compliance to the above secondary side temperature limit is not required during the time that the OSGs are drained for removal from the plant (with the core off-loaded) and the time the RSGs have been installed and may be pressurized (prior to reloading fuel in the core). This administrative LCS change is not adverse, since the OSGs and RSGs do not need to be operable during this SG replacement time frame.

The configuration and/or number of the RSG internal components (tube bundle, feedwater ring, moisture separators, steam dryers) are different than those for the OSG components. These differences also represent functional improvements over the OSG components. The RSGs have a different number of access provision then the OSGs, some of which have also different dimensions. These differences are aimed at improving RSG accessibility for inspections and maintenance.

The RSG lower shell diameter is slightly larger than the OSG diameter (174.6 in. vs. [ ] in.) and the RSG are slightly heavier than the OSGs. However, these differences are within the allowable limits established and allowed for the RSGs.

Based on the above, it is concluded that, the proposed activity does not adversely affect the steam generator ability to:
(1)Function as a part of the RCPB
(2)Transfer heat between RCS and main steam system and
(3)Remove heat from the RCS to achieve and maintain safe shutdown following postulated accidents (other than the large break LOCA).

Based on their physical characteristics, the RSGs provide specific heat transfer capability under defined plant conditions and provide RCPB integrity under defined limits of pressure and temperature. However, replacing OSGs with RSGs does not directly affect any methods of performing or controlling heat transfer or RCPB integrity. The indirect effects of the steam generator replacement on the methods of performing or controlling design functions of other SSCs was evaluated in Sections 3.1 through 3.12 of Appendix A of Reference 1 and it was concluded that the replacement did not adversely affect the ability of performing or controlling any of these design functions. Detailed discussion of these aspects of the replacement is provided in the following paragraphs.

Subsection (a) - RCS Structural Integrity

No. Implementation of steam generator replacement required evaluations and analyses accounting for dead weight, normal operating pressure, seismic, and branch line pipe break (BLPB) loads to demonstrate that structural integrity of the evaluated components was maintained for operation with the RSGs installed. The analyses and evaluations concluded that structural integrity would be maintained.

The number of allowed opening and closing cycles for the primary manway nuts and stud bolts was decreased from 500 to 200. Based on SONGS operating experience over the last 25 years, it was determined that such reduction did not constitute an adverse change. however, as the number of manway opening/closing cycles is not described in the UFSAR, and the design meets the ASME Code criteria.

Operation Details Cont'd:

Subsection (b) - ECCS Performance

No. Reference 2 documents an evaluation of the differences between the RSGs and OSGs which showed that most of the differences are small in magnitude and/or have a negligible impact on ECCS performance. Where the differences are larger or are associated with parameters that have a significant impact on ECCS performance, the differences were found to have an overall beneficial effect on ECCS performance.

Subsection (c) - Non-LOCA Transients

No. The results of analyses and evaluations of UFSAR Chapter 15 non-LOCA transients performed for operation with RSGs are presented in Part 3, Section 3.7 of Reference 1. Reanalyzed events continue to meet acceptance criteria when accounting for RSG characteristics with the performance of mitigating SSCs within Technical Specification limits. For evaluated events, the existing analyses were shown to remain applicable. Events classified as "Not Reanalyzed" are not affected by the RSGs, and events that are "Not Applicable" are not relevant to the non-LOCA transient analyses.

Subsection (d) - Containment P-T Analysis

No. The containment P-T analysis was revised to account for the RSGs. The governing case is the MSLB with MSIV failure, for which peak containment pressure of 56.5 psig for the OSGs decreased to 51.5 psig for the RSGs. The MSLB with MSIV failure peak containment vapor temperature of 409°F for the OSGs decreased to 380°F for the RSGs. Note that the data for the governing case are for the worst case MSLB with MSIV failure case, which changed from the full power case for the OSGs to the zero power case for the RSGs. Although the peak LOCA containment pressure increased from 45.9 psi with the OSGs to 46.7 psi with the RSGs, the peak pressure for the RSGs remains below that for the limiting event (MSLB), which is also below the containment design pressure of 60 psig (see above).

The pressure profiles at 24 hours post-LOCA show that the pressure is below one-half of the peak pressure, thereby confirming compliance with GDC 38. The MSLB event is analyzed for 5000 seconds (1.39 hours) at which time the pressure is well below one-half of the peak pressure, thereby confirming compliance with GDC 38 for MSLB as well. As such, SG replacement is not adverse from the containment P-T analysis perspective.

Subsection (e) - Low Temperature Overpressure Protection

No. Reanalysis of the LTOP transients demonstrated that the existing RCS Technical Specification pressure-temperature limits are met and that the integrity of the RCPB is maintained. The SDCS peak pressure does not exceed the ASME Code allowable. In particular, the reanalysis used the same initial conditions and SDCS relief valve characteristics as described in the UFSAR and analyzed the same LTOP transients. The reanalysis demonstrated that the relieving capacity of the SDCS relief valve remains adequate. Since there were no design changes to the LTOP mitigation equipment, no changes to the LTOP administrative controls, and no changes to operator actions, SG replacement has no effect on how design functions that provide LTOP are accomplished or controlled.

Operation Details Cont'd:

Subsection (f) - RPS, ESFAS, COLSS and CPC

No. SG replacement is independent of, and does not alter, the RPS, ESFAS, COLSS or CPCs, or change the manner in which RPS, ESFAS, COLSS or CPC design functions are performed or controlled.

Subsection (g) - NSSS Performance

No. *For SG replacement, functional and performance requirements for the* RCS, CVCS, SIS, SDCS *and* CSS are met based on existing SSC capabilities. Section 3.10 of Reference 1 addresses that the RCS, CVCS, SIS, SDCS, and CSS will continue to perform their design functions with the RSGs installed. SG replacement does not change the manner in which design functions of the RCS, CVCS, SIS, SDCS and CSS are performed or controlled.

Subsection (h) - *Non-Safety Related Control Systems Performance*

No. The non-safety Control System configuration and setpoints currently used for SONGS Unit 3 are adequate for operation with the RSGs installed. These control systems ensure proper NSSS dynamic response such that the probability for reactor trips is minimized for the analyzed load changes and certain equipment malfunctions, and so that the post-trip responses are acceptable. There are no changes to control systems design or to automatic or manual control capability, so there is no effect on how design functions of these systems are performed or controlled.

Conclusion: Based on the foregoing, changing the OSGs to RSGs does not change an SSC in a manner that adversely affects UFSAR-described design functions or that has an adverse effect on the method of performing or controlling UFSAR-described design functions.

2. Changes a procedure in a manner that adversely affects how UFSAR/DSAR described SSC design function(s) are performed or controlled.

No.
The RSGs provide RCPB integrity under defined limits of pressure and temperature and provide specific heat transfer capability under defined plant conditions based only on their physical characteristics. Procedures will be updated, as necessary, to reflect the RSGs and the updated analyses. The analyses have shown that the RCS and other systems will operate in a manner such that they will continue to perform their design functions. Therefore, steam generator replacement does not involve changing any procedure in a manner that adversely affects how UFSAR-described design functions related to the steam generators are performed or controlled. The effect of the steam generator replacement on procedures pertinent to other SSCs was evaluated in Sections 3.1 through 3.12 of Appendix A of Reference 1. Changes to procedures required to reflect the RSGs will not adversely affect the methods of performing or controlling UFSAR-described functions.

3. Involves revising or replacing a UFSAR/DSAR described method of evaluation that is used in establishing the

Operation Details Cont'd:

design bases or used in the safety analyses in an adverse manner.

Yes.
The stress analyses for the RSGs utilized the ABAQUS computer program, whereas UFSAR Sections 3.9.1.2.2.1.11 and 3.9.1.2.2.2.3 describe these analyses for the OSGs as utilizing the ANSYS computer program.

The RG 1.121 tube wall thinning evaluation for the RSGs changed the methods of evaluation from those described in UFSAR 5.4.2.3.1.3. Instead of using the CEFLASH computer program for MSLB mass-energy release analysis, the peak forces acting on the RSG tubes were calculated manually using the maximum possible differential pressure across the tube wall. In addition, instead of using a two-step process with the STRUDL and ANSYS computer programs to calculate displacement histories and tube stresses, respectively, the tube stresses were determined from blowdown forces using only the ANSYS computer program. Finally, for the RSGs, the tube stresses due to LOCA, DBE and MSLB were combined into one "limiting event" and the stresses for this combined event were calculated, whereas for the OSGs, the analyses considered primary loop branch line pipe break plus DBE and MSLB plus DBE separately.

Subsection (a) - RCS Structural Integrity

No. The original RCS analysis to generate seismic response spectra for RCS branch nozzle and subsystem analysis was performed using the STRUDL computer code to define the structural properties of the RCS coupled to the containment structure, basement and soil. More recent analyses performed in support of Tcold reduction, inoperable steam generator snubbers during Modes 4 and 5, as well as those for steam generator sensitivity studies, used the ANSYS computer code to define the structural properties of the RCS. The equivalence of the two models (ANSYS and STRUDL) was confirmed by comparison of their frequencies and mode shapes. The ANSYS code is an approved code for RCS structural analysis at SONGS, therefore, the change in the method of evaluation from STRUDL to ANSYS is not adverse.

Yes. The seismic analysis of the RVI was performed in accordance with the methodology described in topical report CENPD-178-P, Revision 1-P. UFSAR Section 3.7.3.14.1 references topical report CENPD-178. CENPD-178 was originally submitted to the NRC in 1976 as a Combustion Engineering topical report. Subsequent to submittal of CENPD-178, modeling techniques, computer codes, testing methods and acceptance criteria were modified in response to changes in licensing requirements. As a result, the entire report was revised and resubmitted as CENPD-178-P, Revision 1-P. The use of CENPD-178-P, Revision 1-P, is a revision to the UFSAR-described method of evaluation used in establishing the design bases.

Subsection (b) - ECCS Performance

No. Instead of reanalyzing the UFSAR-described ECCS performance evaluations for LBLOCA, SBLOCA, and post-LOCA long term cooling, an evaluation was performed to assess the effects of revised parameters on the Analyses of Record (AORs) to determine whether the AORs remain valid. This evaluation is documented in Reference 2. The AOR methodologies were not revised or replaced.

Subsection (c) - Non-LOCA Transients

No. The transient simulation code used for analyzing the loss of all normal feedwater flow and loss of all

Operation Details Cont'd:

feedwater flow with a concurrent single failure of an active component was changed from the CESEC computer program (for OSGs) to the CENTS computer program. However, this change is consistent with Reference 3, which approved use of the CENTS computer program for non-LOCA transient analysis for SONGS Units 2 and 3. The CENTS computer program is being used within the constraints and limitations associated with the method; therefore, the change in method of evaluation is not adverse. No other non-LOCA transient methods of evaluation were revised or replaced.

Subsection (d) - Containment P-T Analysis

No. For the RSGs, the blowdown phase of the LOCA mass and energy release was simulated with the CEFLASH 4A computer program consistent with UFSAR Section 6.2.1.3.3. The reflood and post-reflood phases of the LOCA for cold leg breaks were simulated using the FLOOD3 computer code consistent with UFSAR Section 6.2.1.3.4. The long term cooling mass and energy releases were determined with the CONTRANS and COPATTA computer programs. CONTRANS calculated residual heat addition from the primary and secondary metal and SG inventory. The residual heat addition was input into COPATTA. Decay heat was developed consistent with the prior analysis and input directly into COPATTA. This is the same methodology as that described in UFSAR Section 6.2.1.3.5.

Subsequent to the original analysis for the OSGs, the NRC granted SONGS Units 2 and 3 a change in the licensed power limit and an exception from the 10 CFR Part 50, Appendix K, requirement to use a 2 percent power measurement uncertainty in ECCS/LOCA related analyses. The sum of the licensed power limit and the power measurement uncertainty did not change as a result of the NRC action. Consequently, the "analysis power" (licensed power plus power measurement uncertainty) for the decay heat model, as well as the full power M-E and P-T cases, did not change even though the NRC granted an apparent change in the licensed full power level.

For the RSGs, the MSLB containment M-E analysis was performed using the SGNIII computer program, consistent with UFSAR Sections 6.2.1.4 and 6.2.1.4.4. A 2 percent expansion factor was used which is consistent with UFSAR Section 6.2.1.4.3. The mass and energy data was then used to determine the containment response using the COPATTA computer program. This is the same methodology as that described in UFSAR Section 6.2.1.1.3.1.C.

Based on the above, containment P-T re-analyses were performed in accordance with UFSAR-described methods of evaluation.

Subsection (e) - Low Temperature Overpressure Protection

No. The UFSAR does not describe a method of evaluation for analyzing LTOP transients.

Subsection (f) - RPS, ESFAS, COLSS and CPC

No. The methodology described in report CEN 112(S), Rev. 0, cited in UFSAR Section 7.2.11.2 for calculating setpoints, including uncertainties, was used in evaluating RSG effects on setpoints and uncertainties.

Operation Details Cont'd:

Subsection (g) - NSSS Performance

No. RSG effects on the NSSS systems were assessed using qualitative evaluations, manual calculations and the results of various analyses and evaluations. These evaluations, calculations and analyses do not involve revising or replacing a UFSAR-described method of evaluation that was used in establishing the design bases or used in the safety analyses.

Subsection (h) - Non-Safety Related Control Systems Performance

No. The methods of evaluation for non-safety control system performance are not described in the UFSAR.

Based on the above, it is concluded that the computer programs used for stress analysis of the RSGs and for seismic analyses of the reactor vessel internals were changed. These changes constitute a revision to the methods of evaluation described in UFSAR 3.7.3.14.1, 3.9.1.2.2.1.11, 3.9.1.2.2.2.3 and 5.4.2.3.1.3.

4. Involves a test or experiment not described in the UFSAR/DSAR, where an SSC is utilized or controlled in a manner that is outside the reference bounds of the design for that SSC, or is inconsistent with the analysis or descriptions in the UFSAR/DSAR.

No.
The proposed activity involves functional testing of the RSGs after installation. The testing will be performed during normal operation of the plant when the RSGs and the associated SSCs will be utilized strictly within the bounds of their design and consistent with the approved plant operating procedures. The testing will involve only recording of the selected plant operating parameters in accordance with the approved test procedure, using the instruments installed in the plant. The effect of RSG installation on requiring tests and experiments involving other SSCs was evaluated in Sections 3.1 through 3.12 of Appendix A of Reference 1. The RSGs do not require any tests or experiments not described in the UFSAR, where an SSC would be utilized or controlled in a manner that would be outside the bounds of the design, or inconsistent with the analyses of record for that SSC.

5. Requires a change to the Technical Specifications.

Yes.
Steam generator replacement results in a change in tubing material (from Alloy 600MA to Alloy 690TT) and wall thickness (from 0.048 in. to 0.0429 in.). The new tubing was evaluated in accordance with Regulatory Guide 1.121 in order to determine new tube plugging criteria. For OSGs, Technical Specification 5.5.2.11 requires that tubes be plugged or repaired if in-service inspection determines that the tube contains flaws to a depth equal to, or greater than, 44% of the nominal tube wall thickness. The RSG tubes will be required to be plugged if they contain flaws to a depth equal to, or greater than, 35% of the nominal wall thickness.

Therefore, the proposed activity involves changes to Technical Specifications 3.4.17, "SG Tube Integrity," 5.5.2.11, "SG Program," and 5.7.2, "Special Reports."

Operation Details Cont'd:

Subsection (a) - RCS Structural Integrity

No. SG replacement required assessing its effect on RCS components. The assessment concluded that no Technical Specifications were affected.

Subsection (b) - ECCS Performance

No. The Technical Specifications that apply to the ECCS system and analyses, 3.5, "Emergency Core Cooling Systems," and 5.7.1.5, "Core Operating Limits Report (COLR)," were reviewed for impact and do not require changes.

Subsection (c) - Non-LOCA Transients

No. Review of the Technical Specifications revealed one change required for the non-LOCA transient analyses and evaluations. License amendments 212 (Unit 2) and 204 (Unit 3) established revised high power level trip values in Technical Specification 3.7.1 for three and four MSSVs out of service per steam generator. The revised trip values are consistent with the values documented in Part 3, Section 3.7 of Reference 1. Although revised Technical Specification 3.7.1 also increases the allowable power level for one and two MSSVs inoperable per steam generator, this aspect of the Technical Specification change is not required by and will not be impacted by the RSGs. Hence, SG replacement does not require any Technical Specification changes for non-LOCA transients.

Subsection (d) - Containment P-T Analysis

Yes. As discussed above in II.1 (d), the peak containment pressure of 56.5 psig for the OSGs decreases to 51.5 psig for the RSGs. Therefore, the proposed activity does involve a change to Technical Specification 5.5.2.15, "Containment Leakage Rate Test Program," (and its corresponding Bases) and LCS 3.6.100, "Pre-Stressed Concrete Containment Tendon Surveillance Program."

Subsection (e) - Low Temperature Overpressure Protection

No. SG replacement required reanalysis of the LTOP mass and energy transients but no Technical Specification changes.

Subsection (f) - RPS, ESFAS, COLSS and CPC

No. SG replacement does not involve any changes to the Technical Specifications pertinent to the RPS, ESFAS, COLSS or CPCs.

Subsection (g) - NSSS Performance

No. SG replacement does not involve any changes to the Technical Specifications pertinent to the RCS, CVCS, SIS, SCS, or CSS.

Operation Details Cont'd:

Subsection (h) - Non-Safety Related Control Systems Performance

No. The Technical Specifications do not include non-safety control system setpoints or performance requirements. As a result, the proposed activity does not require changes to any Technical Specifications.

Based on the above, it was concluded that the proposed activity involves changes to Technical Specification 3.4.17, "Steam Generator (SG) Tube Integrity," 5.5.2.11, "Steam Generator Tube Surveillance Program," 5.7.2, "Special Reports,"and TS 5.5.2.15, "Containment Leakage Rate Testing Program," as well as to LCS 3.7.109, "Steam Generator Pressure/Temperature Limitation".

The aspect of the proposed activity requiring changes to the Technical Specifications are addressed in PCN-583 submitted for NRC approval.

SECTION III: SCREEN CONCLUSION (See Keypoints Section III)

Based on the responses provided in Section II, the proposed activity is:

____ NOT adverse

__X__ Adverse, record the 50.59 Evaluation (SE) notification and task: 800175664-0170

The 50.59 screen concern that requires evaluation is Screen Concern 3. Refer to Section II.3 and Subsection II..3(a) for further information.

Specifically, the following issues have to be evaluated:

1. The OSG structural integrity analyses described in UFSAR Sections 3.9.1.2.2.1.11 and 3.9.1.2.2.2.3 utilized the ANSYS computer program, whereas the RSG analyses utilized the ABAQUS computer program.

2. The seismic analysis of the RVI for the RSGs was performed in accordance with the methodology described in topical report CENPD-178-P, Revision 1-P. The OSG analyses described in UFSAR Section 3.7.3.14.1 reference CENPD-178.

3. The RG 1.121 tube wall thinning evaluation for the RSGs used methods of evaluation different from those described in UFSAR 5.4.2.3.1.3.

(a)The OSGs used the computer program CEFLASH for MSLB mass-energy release analysis, the RSG analyses used manual calculation methods using the maximum differential pressure across the tube wall.
(b)Instead of using a two-step process with the computer programs STRUDL and ANSYS to calculate displacement histories and tube stresses, respectively, tube stresses were determined from blowdown forces using only ANSYS.
(c)For the RSGs, the LOCA, DBE and MSLB events were combined into one "limiting event" and the stresses for this combined event were calculated, whereas the OSG analyses considered LOCA plus DBE [UFSAR

Operation Details Cont'd:

5.4.2.3.1.3.A] and MSLB plus DBE [UFSAR 5.4.2.3.1.3.B] as separate events.

Technical Specification changes are required as a result of the proposed activity. The Technical Specification changes identified in this screen are included in PCN-583 sent for NRC#s approval, andare hence excluded from the 50.59 Evaluation of the proposed activity. NRC approval of PCN-583 is a prerequisite to closing the NECP governing this proposed activity

Updated Final Safety Evaluation Report (UFSAR) changes are required as a result of the proposed activity. Specifically, the UFSAR sections listed below are affected by the proposed activity and are revised accordingly. Prior NRC approval is either being sought through PCN-583, or is not required as demonstrated in this screen. The changes are "Pending" NRC approval of PCN-583 and subsequent closure of NECP 800071702 (Unit 2 RSG) and 800071703 (Unit 3 RSG).

3.7 Seismic Design	change # 2008-38
3.9 Mechanical Systems and Componentschange # 2008-10	
3.11Environmental Design (P-T Analysis)	change # 2008-11
4.4 Thermal and Hydraulic Design	change # 2008-12
5.1 Summary description	change # 2008-13
5.2 Integrity of RC Pressure Boundary	change # 2008-14
5.4 Component and Subsystem Design	change # 2008-15
6.2 Containment systems	change # 2008-16
6.3 Emergency Core Cooling System	change # 2008-17
7.2 Reactor Protective System	change # 2008-18
9.3 Process Auxiliaries	change # 2008-19
10.1Summary description	change # 2008-20
10.3 Main Steam Supply System	change # 2008-21
15.10Updated accident Analyses	change # 2008-33

SECTION IV: REFERENCES (See Keypoints Section IV)

1.WCAP-16811-P, #SONGS Units 2and 3 Replacement Steam Generator Project, NSSS Licensing Topical Report."
2.CN-OA-05-71, "Evaluation of Impact of RSGs on the SONGS Units 2 and 3 ECCS Performance Analysis."
3.Letter from H.N. Berkow (NRC) to G. Bischoff (Westinghouse Electric Company), November 24, 2004, Subject: Final Safety Evaluation for Topical Report WCAP-15996-P, "Technical Description Manual for CENTS Code."

UFSAR Sections Reviewed:

1	Introduction and General Description of Plant
1.2.3	Nuclear Steam Supply System (NSSS)

Operation Details Cont'd:

3. Design of Structures, Components, Equipment and Systems
3.7 Seismic Design
3.9 Mechanical Systems and Components
App. 3.9A Dynamic System Analyses of Reactor Coolant System, Reactor Internals ECCS Piping, and CEDMs Under Faulted Conditions
3.11 Environmental Design of Mechanical and Electrical Equipment
4 Reactor
4.4 Thermal and Hydraulic Design
5 Reactor Coolant and Connected Systems
5.1 Summary Description
5.2 Integrity of Reactor Coolant Pressure Boundary (RCPB)
5.4 Component and Subsystem Design
6 Engineered Safety features
6.2 Containment Systems
6.3 Emergency Core Cooling System
7 Instrumentation and Controls
7.2 Reactor Protective System
7.3 Engineered Safety Features System
7.7 Control Systems not Required for Safety
8 Electric Power
8.3 Onsite Power Systems
9 Auxiliary Systems
9.3 Process Auxiliaries
10 Steam and Power Conversion Systems
10.1 Summary Description
10.3 Main Steam Supply System
10.4 Other Features of Steam and Power Conversion System
15 Accident Analyses
15.6 Decrease in Reactor Coolant Inventory
15.10 Updated Accident Analyses
App. 15.D Responses to IE Bulletins 79 27 and 80 06, IE Information Notice 79 22, and Multiple Control System Failure Evaluation

Technical Specifications Reviewed:
3.1.9 Boration Systems # Operating
3.1.10 Boration Systems # Shutdown
3.3.1 Reactor Protective System (RPS) Instrumentation # Operating
3.3.2 Reactor Protective System (RPS) Instrumentation # Shutdown
3.3.5 Engineered Safety Features Actuation System (ESFAS) Instrumentation
3.4 Reactor Coolant System
3.4.3 RCS Pressure and Temperature (P/T) Limits
3.4.4 RCS Loops # Modes 1 and 2
3.4.5 RCS Loops # Mode 3
3.4.6 RCS Loops # Mode 4
3.4.7 RCS Loops # Mode 5, Loops Filled

Operation Details Cont'd:

3.4.12.1 Low Temperature Overpressure Protection (LTOP) System, RCS Temperature d
PTLR Limit
3.4.12.2 Low Temperature Overpressure Protection (LTOP) System, RCS Temperature >
PTLR Limit
3.4.13 RCS Operational Leakage
3.4.17 RCS Steam Generator (SG) Tube Integrity
3.5 Emergency Core Cooling Systems (ECCS)
3.6.6.1 Containment Spray and Cooling Systems
3.7.1 Main Steam Safety Valves (MSSVs)
3.7.2 Main Steam Isolation Valves (MSIVs)
3.7.3 Main Feedwater Isolation Valves (MFIVs)
3.7.5 Auxiliary Feedwater (AFW) System
3.8.1 AC Sources # Operating
3.9.4 Shutdown Cooling (SDC) and Coolant Circulation # High Water Level
3.9.5 Shutdown Cooling (SDC) and Coolant Circulation # Low Water Level
5.5.2.11 Steam Generator (SG) Program
5.5.2.15 Containment ILRT
5.7.1.5 Core Operating Limits Report (COLR)
5.7.1.6 Reactor Coolant System (RCS) Pressure and Temperature Limits Report
(PTLR)

Licensee Controlled Specifications Reviewed:
3.7.109 Steam Generator Pressure/Temperature Limitation

SECTION V: SUMMARY (See Keypoints Section V)

Three corrections are made to this screen as compared to the previous screen (Operation 800175664-0030).

1. End of Section II: Changed to LCS 3.7.109, "Steam Generator Pressure/Temperature Limitation"

instead of LCS 3.6.100, "Pre-stressed Concrete Containment Tendon Surveillance Program".

2. Beginning of Section III; The operation of the 50.59 Evaluation (SE) should be 800175664-0170.

3. In Section III Item 3, the UFSAR Change # is 2008-38 instead of 2008-09 for UFSAR Section 3.7 Seismic Design.

The conclusions of the original screen remain valid.

ENCLOSURE 8

Unit 3 – 10 CFR 50.59 Evaluation
Completed On: 07/31/2009

(Non-Proprietary)

EDISON

SONGS

Operations Detail

Order: NECP 800175664
Description: Steam Generator Replacement Mstr ECP U3

Operation No: 0170 **Sub Op:** **Op Wrkctr:** ED_SGD SGRP Design
Description: 10CFR50.59 Evaluation Unit 3 RSG Lead NE
Work: 4.0 H **Number:** 1 **Duration:** 4.0 H
Risk level: **Potential Rx Impact:**
Risk Type:
RiskComments:

Operation Details:

10CFR50.59 Evaluation Unit 3 RSG Lead NECP
Completing a 50.59 Evaluation requires PQS T3EN99

Activity Title:

Steam Generator Replacement - Unit 3

Activity Description:

The proposed activity replaces the design disclosure documentation and reference documentation for the SONGS Unit 3 Original Steam Generators (OSGs) with that for the Replacement Steam Generators (RSGs) and performs functional testing of the RSGs. This replacement and testing is required as a result of physically replacing the OSGs with RSGs. Having the OSGs replaced with the RSGs will improve the efficiency and reliability of Unit 3 by replacing a large number of plugged or otherwise degraded heat transfer tubes in each OSG with new tubes made from thermally-treated Alloy 690, which is less susceptible to degradation than the mill annealed Alloy 600 material used for OSG heat transfer tubing. Replacement of the steam generators is a replacement in-kind in terms of an overall fit, form, and function with no, or minimal, permanent modifications to the plant systems, structures, and components (SSCs).

The proposed activity covered by this evaluation includes the following:

1. Replacing the Original Steam Generator (OSG) documentation with the Replacement Steam Generator (RSG) documentation. This includes voiding/superseding the original Combustion Engineering/Westinghouse Electric Co. (CE/WEC) documents, as appropriate, and adding the new Mitsubishi Heavy Industries (MHI) design documents to the database.
2. Revising affected design bases documents (DBDs, UFSAR, calculations, etc.).
3. Revising other affected documents (90000 series documents, EQDPs, System Descriptions, Licensee Controlled Specifications (LCS), etc.).
4. Functional (performance) testing of the RSGs after installation.

Note: Activities associated with the physical removal of the OSGs and installation of the RSGs, such as creation and restoration of a temporary containment building construction opening, replacement of the steam generator insulation, re-calibration the steam generator level transmitters, or Cycle 16 Reload Analysis, are not addressed

Operation Details Cont'd:

herein, but instead are addressed in the specific NECPs governing these activities.

A 10 CFR 50.59 Screen for the proposed activity determined that it involved changes to UFSAR-described methods of evaluation. These changes to methods of evaluation are summarized below.

1. The OSG stress analyses described in UFSAR Sections 3.9.1.2.2.1.11 and 3.9.1.2.2.2.3 utilized the ANSYS computer program, whereas the RSG analyses utilized the ABAQUS computer program.

2. The seismic analysis of the RVI for the RSGs was performed in accordance with the methodology described in Topical Report CENPD-178-P, Revision 1-P. The corresponding OSG analyses, described in UFSAR Section 3.7.3.14.1, reference Topical Report CENPD-178.

3. The RG 1.121 tube wall thinning evaluation for the RSGs changed the methods of evaluation from those described for the OSGs in UFSAR 5.4.2.3.1.3, as follows:

(a) The OSG analysis used the CEFLASH computer program for MSLB mass-energy blowdown analysis; the RSG analyses used manual calculation methods using the maximum differential pressure across the tube wall during the MSLB.
(b) The OSG LOCA analysis contained a two-step process utilizing the STRUDL and ANSYS computer programs to calculate displacement histories and tube stresses, respectively. The corresponding RSG analysis determined tube stresses from blowdown forces using only the ANSYS computer program.
(c) The OSG analyses considered primary loop branch line pipe break plus DBE [UFSAR 5.4.2.3.1.3.A] and Main Steam Line Break (MSLB) plus DBE [UFSAR 5.4.2.3.1.3.B] as separate events for determining tube stress. For the RSGs, the LOCA, DBE, and MSLB events were combined into one "limiting event" and the stresses for this combined event were calculated.

Summary of Evaluation:

A departure from a method of evaluation described in the UFSAR used in establishing the design bases or in the safety analyses, means changing any of the elements of the method described in the UFSAR, unless the results of the analysis are conservative or essentially the same.

RCS Structural Integrity Changing from ANSYS to ABAQUS

UFSAR Section 3.9.1.2.2.1.11 describes ANSYS as a large-scale, general-purpose, finite element program for linear and nonlinear structural and thermal analysis of reactor coolant loop components. The RCS structural integrity analyses for RSGs utilized ABAQUS in lieu of ANSYS. Like ANSYS, ABAQUS is a large-scale, general-purpose finite element program for linear and nonlinear structural and thermal analysis. In using ABAQUS, elements of the approved OSG method of evaluation were changed, but the analyses did not use an entirely new method altogether.

ABAQUS and ANSYS were compared, using thermal and stress sample problems. The results of these sample

Operation Details Cont'd:

analyses demonstrated that in all cases the ANSYS and ABAQUS results varied from theoretical solutions by no more than 1%, and ABAQUS and ANSYS results themselves were also within 1% of each other. A difference of 2% is considered within the margin of error for this type of analysis; hence ABAQUS produced results that are essentially the same as ANSYS, and does not constitute a departure from a method of evaluation described in the UFSAR used in establishing the design basis or in the safety analyses.

Seismic Analysis of Reactor Vessel Internals Changing from Topical Report CENPD-178 to CENPD-178-P, Revision 1-P

The original seismic analysis of the SONGS Unit 2 and 3 RVI (for the OSGs) was performed with the methodology described in C-E Topical Report CENPD-178 as referenced in UFSAR Section 3.7.3.14.1. Subsequent to submittal of CENPD-178, C-E modified modeling techniques, computer codes, testing methods, and acceptance criteria in response to changes in licensing requirements. As a result, the entire report was revised and resubmitted to the NRC as CENPD-178-P, Revision 1-P. CENPD-178-P, Revision 1-P, was approved by the NRC in NRC Letter, H. Bernard to A. Scherer, "Acceptance for Referencing of Topical Report CENPD-178 (P)," dated August 6, 1982.

Westinghouse, the current owner of the methodology, used it to perform the RSG RVI seismic analyses, consistent with its intended application, constraints and limitations. Consequently, its use does not constitute a departure from a method of evaluation described in the UFSAR used in establishing the design bases or in the safety analyses.

Tube Wall Thinning Analyses Changing from CEFLASH, STRUDL and ANSYS to Manual Calculations and ANSYS

The OSG analysis used the CEFLASH computer program for MSLB mass-energy blowdown analysis, whereas the RSG analysis used manual calculations to represent the MSLB blowdown loads by applying the maximum possible tube differential pressure. This manual calculation approach applies a differential pressure on the RSG tube wall that bounds the pressure calculated by CEFLASH.

Instead of using a two-step process for the OSGs LOCA analysis, utilizing computer programs STRUDL and ANSYS to calculate displacement histories and tube stresses, respectively, tube stresses for RSGs were determined from blowdown forces using only ANSYS. This use of ANSYS is consistent with its intended application, constraints and limitations as described in UFSAR 3.9.1.2.2.1.11, and, therefore, does not represent a departure from a method of evaluation.

For RSGs , the LOCA, DBE and SLB events were combined as one "limiting event" and the tube stresses were calculated for this event, whereas the OSG analyses considered PLPB plus DBE [UFSAR 5.4.2.3.1.3.A] and MSLB plus DBE [UFSAR 5.4.2.3.1.3.B] separately. Since the RSG approach combines the loads of three independent events (LOCA + DBE + SLB), rather than analyzing PLPB + DBE and MSLB + DBE loads separately, the RSG method of evaluation is conservative relative to the OSG method of evaluation.

The results of the RSG tube wall thinning analysis are conservative or essentially the same as results from the UFSAR-described tube wall thinning analysis for the OSGs. In addition, this approach results in a lower tube wall plugging limit (44% for OSGs versus 35% for RSGs) which is also conservative. Therefore, the method of

Operation Details Cont'd:

evaluation change for the tube wall thinning analysis is not a departure from a method of evaluation described in the UFSAR used in establishing the design bases or in the safety analyses.

Based upon the results of this evaluation:

__X__ Implement the activity per plant procedures without obtaining a License Amendment.

______ Request and receive a License Amendment prior to implementation.

10CFR50.59 Evaluation

Completing a 50.59 Evaluation requires PQS T3EN99

SO123-XV-44.1 should be used to determine the content of each response (see Section 6.2 for additional guidance).

If the answer to any of the questions is "YES", then the proposed activity may not be implemented until a License Amendment has been obtained from the NRC.

UFSAR refers to the current FSAR, approved changes to the FSAR which have not yet been submitted to the NRC by amendment and documents included in the FSAR by reference.

1. Does the proposed activity result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the UFSAR/DSAR? (see 6.2.1)

N/A

2. Does the proposed activity result in more than a minimal increase in the likelihood of occurrence of a malfunction of an SSC important to safety previously evaluated in the UFSAR/DSAR? (see 6.2.2)

N/A

3. Does the proposed activity result in more than a minimal increase in the consequences of an accident previously evaluated in the UFSAR/DSAR? (see 6.2.3)

N/A

4. Does the proposed activity result in more than a minimal increase in the consequences of a malfunction of an SSC important to safety previously evaluated in the UFSAR/DSAR? (see 6.2.4)

N/A

5. Does the proposed activity create a possibility for an accident of a different type than any previously evaluated in the UFSAR/DSAR? (see 6.2.5)

Operation Details Cont'd:

N/A

6. Does the proposed activity create a possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated in UFSAR/DSAR? (see 6.2.6)

N/A

7. Does the proposed activity result in a design basis limit for a fission product barrier as described in the UFSAR/DSAR being exceeded or altered? (see 6.2.7)

N/A

8. Does the proposed activity result in a departure from a method of evaluation described in the UFSAR/DSAR used in establishing the design bases or in the safety analyses? (see 6.2.8)

No.

The SONGS 50.59 Resource Manual (Procedure SO123-XV-44.1), states that departure from a method of evaluation described in the UFSAR used in establishing the design bases or in the safety analyses, means changing any of the elements of the method described in the UFSAR, unless the results of the analysis are conservative or essentially the same. One example of a change that is not considered a departure from a method of evaluation described in the UFSAR is:

"Use of a methodology revision that is documented as providing results that are essentially the same as, or more conservative than, either the previous revision of the same methodology or another methodology previously accepted by NRC through issuance of an SER."

The 50.59 Resource Manual continues:

"Analytical results obtained by changing any element of a method are conservative relative to the previous results, if they are closer to design bases limits or safety analyses limits (e.g., applicable acceptance guidelines).

"For example, a change from 45 psig to 48 psig in the result of a containment peak pressure analysis (with design basis limit of 50 psig) using a revised method of evaluation would be considered a conservative change when applying this criterion. In other words, the revised method is more conservative if it predicts more severe conditions given the same set of inputs. This is because results closer to limiting values are considered conservative in the sense that the new analysis result provides less margin to applicable limits for making potential physical or procedure changes without a license amendment."

RCS Structural Integrity Changing from ANSYS to ABAQUS

ANSYS was the computer program used to perform the OSG RCS structural integrity analyses. UFSAR Section 3.9.1.2.2.1.11 describes ANSYS as a large-scale, general-purpose, finite element program for linear and nonlinear structural and thermal analysis of reactor coolant loop components.

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In evaluating the impact of the RSGs, the RCS structural integrity analyses utilized the ABAQUS computer program. Like ANSYS, ABAQUS is a large-scale, general-purpose finite element program for linear and nonlinear structural and thermal analysis. In using ABAQUS in lieu of ANSYS for these analyses, elements of the approved OSG method of evaluation were changed, but the analyses did not use an entirely new method altogether.

Only linear elastic elements of ABAQUS for were utilized the RSG analyses. One analysis, for the tube-to-tubesheet seal weld, L5-04GA431 (SO23-617-1-C1306), includes an ABAQUS model that contains gap elements to account for boundary conditions above the seal weld. When gap elements are used in a model combined with time history analysis of dynamic loading, they can have an energy dissipating effect that changes the results (if the gaps open and close in the response time history). If loads are applied as dynamic loads, these would represent nonlinear aspects of the analysis.

In the tube-to-tubesheet seal weld analysis, the discussion and tables of transient loads, which are primarily thermal loads, should not be taken to imply that any loads were applied as dynamic loads. These tables were used only to find the maximum temperature difference to apply statically to the model.

Essentially, the gap elements were used as a convenience that replaced having to manually iterate the model to perform a displacement compatibility check (i.e., confirm that displaced elements do not physically overlap). This practice is well-known and accepted in the finite element analytical arena, and is still well within the linear elastic range of modeling. In essence, one can get the same exact results by using a computer program without gap element capability (e.g., a simple linear elastic computer program), and manually iterating node connectivity so that the displacements are compatible.

Additionally, the analyses of the tube bundle assembly did not model spaces between tubes and anti-vibration bars as gap elements. These spaces were either too small or were conservatively considered to be too large to interact with a tube (thereby providing no tube support). As such, the RSG analyses used only linear elastic analysis capabilities of ABAQUS.

To demonstrate that this element change does not represent a departure from a method of evaluation, one must compare the results of sample runs made using each program and demonstrate that the new program yields results that are conservative or essentially the same as those obtained using the UFSAR-described program. ABAQUS and ANSYS were compared, using sample thermal analysis and stress analysis problems. The analyzed structures were a thick-walled cylinder, a thick-walled sphere, and a curved beam of varying dimensions. The cylinder model included 2-D axisymmetric, 3-D shell, and 3-D solid elements. The sphere model included 2-D axisymmetric and 3-D solid elements. The thermal analysis problem included steady-state and non steady-state loads. The stress analysis problem included internal pressure, thermal, and concentrated loads. Loading conditions varied, but were applied to the corresponding structural model consistently. Theoretical solutions were also derived for comparison purposes.

The results of these sample analyses (contained in MHI document number KAS-200800014 [SO23-617-1-M1379]) were compared to each other, as well as to the theoretical solutions. In all cases, the ANSYS and ABAQUS results varied from the theoretical solutions by less than 1%, and ABAQUS and ANSYS results were also within 1% of each other. A difference of 2% is considered within the margin of error for this type

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of analysis; hence, ABAQUS produces results that are essentially the same as those produced by ANSYS. Therefore, the use of the ABAQUS computer program for the RCS structural analysis does not constitute a departure from a method of evaluation described in the UFSAR used in establishing the design basis or in the safety analyses.

Seismic Analysis of Reactor Vessel Internals Changing from Topical Report CENPD-178 to CENPD-178-P, Revision 1-P

The original seismic analysis of the SONGS Unit 2 and 3 RVI was performed in accordance with the methodology described in Topical Report CENPD-178, as referenced in UFSAR Section 3.7.3.14.1. CENPD-178 was originally submitted to the NRC in 1976 as a Combustion-Engineering Topical Report. Subsequent to submittal of CENPD-178, C-E modified modeling techniques, computer codes, testing methods, and acceptance criteria in response to changes in licensing requirements. As a result, the entire report was revised and resubmitted to the NRC as CENPD-178-P, Revision 1-P. CENPD-178-P, Revision 1-P, was approved by the NRC in NRC Letter from H. Bernard to A. Scherer, "Acceptance for Referencing of Topical Report CENPD-178 (P)," dated August 6, 1982.
Westinghouse, the current owner of the methodology, used it to perform the RVI seismic analyses for RSGs, consistent with its intended application, and constraints and limitations. Consequently, its use does not constitute a departure from a method of evaluation described in the UFSAR used in establishing the design bases or in the safety analyses.

Tube Wall Thinning Analyses Changing from CEFLASH, STRUDL and ANSYS to Manual Calculations and ANSYS

The R.G. 1.121 tube wall thinning analysis methodologies applied to the OSGs are described in UFSAR Section 5.4.2.3.1.3. The corresponding analysis for the RSGs involved changes to methods of evaluation.

OSG Analysis

The UFSAR-described methodology considered two separate loading conditions, denoted "PLPB + DBE" and "MSLB + DBE," which consisted of two separate combinations of pressure, PLPB (Primary Loop Pipe Break or LOCA), DBE (Design Basis Earthquake), and MSLB (Main Steam Line Break) loadings.

UFSAR Section 5.4.2.3.1.3.A describes the PLPB + DBE analysis performed for the OSGs. The CEFLASH computer program was used to perform the OSG hydraulic dynamic analysis of the primary loop during the PLPB event. CEFLASH produced load time histories on the tube bundle. From these load time histories, the computer program STRUDL was used to calculate tube displacement time histories. Next, peak tube stresses from the PLPB loads were calculated, using the computer program ANSYS, and then combined with DBE tube stresses using the square-root-of-the-sum-of-the-squares (SRSS) method. Finally, tube stresses due to pressure load were added to the combined PLPB and DBE stresses.

The same method was used for the MSLB + DBE analysis, as described in UFSAR Section 5.4.2.3.1.3.B. CEFLASH was used for the hydraulic dynamic analysis of the secondary loop during a postulated MSLB accident. Tube displacement time histories at various elevations in the SG were calculated using STRUDL. ANSYS was then used to calculate tube stresses. DBE stresses were calculated and combined with MSLB

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stresses by SRSS, then added to pressure stresses.

The results of the PLPB + DBE and the MSLB + DBE analyses were compared and tube stresses for the governing condition (PLPB + DBE) were compared to the ASME Code allowables .

RSG Analysis

The R.G. 1.121 tube wall thinning analysis for the RSGs considered a single limiting combination of design basis events: LOCA, DBE and MSLB. CEFLASH was originally used to perform the hydraulic dynamic analysis of the primary loop during a LOCA event to determine dynamic structural response of the steam generators (OSGs) to the impulsive loading imparted by the escaping fluid.

For the RSGs, a structural model of the tube bundle was generated using the ANSYS computer code in the seismic and LOCA rarefaction analyses. New seismic response spectra were developed for the RSGs. The rarefaction wave analysis utilized the OSG pressure-time history (dynamic loadings) developed for the OSGs using CEFLASH. For stress calculations, the maximum seismic and LOCA rarefaction wave loads were combined using the SRSS method.

The MSLB event was conservatively modeled assuming the primary side at the design pressure (the pressurizer safety valve set point), plus 3% accumulation, and the secondary side at atmospheric pressure. The stresses due to MSLB differential pressure in the tubes were added directly to the combined stresses calculated for the seismic and LOCA loads. Thus, the RSG analysis considered LOCA loads, DBE loads, and maximum MSLB primary-to-secondary differential pressure across the tubes simultaneously, which is conservative.

Comparison of OSG to RSG Methods

The OSG and RSG analyses used the same computer program, CEFLASH, for hydraulic dynamic analysis of the primary loop during the primary loop pipe break event..

Instead of using the UFSAR-described two-step process with the STRUDL and ANSYS computer programs to calculate displacement histories and tube stresses, respectively, the RSG analysis determined tube stresses from blowdown forces using only ANSYS. ANSYS is described in UFSAR Section 3.9.1.2.2.1.11 as a large-scale, general-purpose, finite element program for linear and nonlinear structural and thermal analysis of reactor coolant loop components. This use of ANSYS for RSGs is consistent with its intended application, and constraints and limitations as described in UFSAR 3.9.1.2.2.1.11. Therefore, utilizing only ANSYS instead of STRUDL and ANSYS is not a departure from the UFSAR-described method of evaluation.

In contrast to the OSG MSLB + DBE analysis, which used the computer program CEFLASH for the mass-energy blowdown analysis portion, the RSG analysis did not include dynamic hydraulic analysis of the MSLB. Instead, the peak secondary side blowdown forces were conservatively calculated by manually adding the loads resulting from the maximum primary to secondary differential pressure across the tube walls to the LOCA + DBE loads. This manual calculation approach applies a differential pressure on the tube wall that bounds the differential tube wall pressure calculated by CEFLASH.

Finally, for the RSGs, the tube stresses due to LOCA, DBE and MSLB were combined as one "limiting event,"

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whereas the OSG analyses considered PLPB + DBE and MSLB + DBE separately. By inspection, the RSG approach is conservative relative to the OSG method of evaluation.

This conservatism is further apparent by comparing the allowable tube wall plugging criteria contained in Technical Specification 5.5.2.11 for the OSGs (44%) to the criteria proposed in PCN-583 for the RSGs (35%). Clearly, the analysis method used for the RSGs results in a lower plugging limit, which is conservative.

Based on the above, it can be concluded that the results of each element of the RSG tube wall thinning analysis are conservative or essentially the same as results from the UFSAR-described tube wall thinning analysis for the OSGs. Therefore, the RSG tube wall thinning analysis is not a departure from a method of evaluation described in the UFSAR used in establishing the design bases or in the safety analyses.

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(NSG Review Comments)
Document below any issues wherein the NSG reviewer either disagrees with the response to a question, or where there is inadequate detail provided to support the conclusion:

Does the change require prior NRC approval?