Nuclear Operating Company

South Texas Project Electric Generating Station P.O. Box 289 Wadsworth, Texas 77483

May 12, 2003
NOC-AE-03001506
10CFR50.59
File: G25

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
One White Flint North
11555 Rockville Pike
Rockville, MD 20852

South Texas Project
Units 1 & 2
Docket Nos. STN 50-498, STN 50-499
<u>10CFR50.59 Summary Report</u>

Pursuant to the requirements of 10CFR50.59, the attached report contains a brief description and summary of the 10CFR50.59 evaluations of changes, tests and experiments conducted at the South Texas Project.

This report includes six evaluations that were omitted from two previous reports. These evaluations are noted in the attachment. This was documented in the corrective action program with actions in place to address the issue. An extensive review was completed to ensure no other evaluations were omitted.

If there are any questions regarding this summary report, please contact Ms. K. A. Work at (361) 972-7936 or me at (361) 972-7136.

Scott M. Head

Scott M. Head
Manager, Licensing

kaw

Attachment: 10CFR50.59 Evaluation Summaries

A001

cc:

(paper copy)

Ellis W. Merschoff
Regional Administrator, Region IV
U.S. Nuclear Regulatory Commission
611 Ryan Plaza Drive, Suite 400
Arlington, Texas 76011-8064

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
One White Flint North
11555 Rockville Pike
Rockville, MD 20852

Richard A. Ratliff
Bureau of Radiation Control
Texas Department of Health
1100 West 49th Street
Austin, TX 78756-3189

Cornelius F. O'Keefe
U. S. Nuclear Regulatory Commission
P. O. Box 289, Mail Code: MN116
Wadsworth, TX 77483

C. M. Canady
City of Austin
Electric Utility Department
721 Barton Springs Road
Austin, TX 78704

(electronic copy)

A. H. Gutterman, Esquire
Morgan, Lewis & Bockius LLP

L. D. Blaylock
City Public Service

Mohan C. Thadani
U. S. Nuclear Regulatory Commission

R. L. Balcom
Texas Genco, LP

A. Ramirez
City of Austin

C. A. Johnson
AEP Texas Central Company

Jon C. Wood
Matthews & Branscomb

Summaries of the following 10CFR50.59 Evaluations are provided in this attachment:

1.	95-4416-149	ESF status monitoring conversion to Integrated Computer System (ICS)
2.	96-2014-95	Unit 1 ERFDADS monitor replacement
3.	97-0021	Technical Specification Basis RCS pressure for LTOP revision
4.	97-0025	Human interface modification in the Main Control Room
5.	97-6297-7	Engineering evaluation for abandonment of 150 ton essential chillers
6.	97-15832-4	UFSAR change regarding the 150 ton essential chiller abandonment
7.	98-0001	Addition of HEPA filter to RCB Supplemental Purge Exhaust System
8.	98-19444-19	Large bore pipe change associated with the Unit 2 steam generator replacement
9.	98-19445-6	Removal and reinstallation of the steam generator blowdown system associated with the Unit 2 steam generator replacement
10.	99-66-23	Reanalysis of the non-LOCA safety evaluation associated with the steam generator replacement program
11.	99-66-64	Replacement of the LOCA/MSLB pressure/temperature analyses associated with the steam generator replacement program
12.	99-2157-2	Revision of UFSAR section 5.4.2.1.3 to allow for possibility of small amount of copper-bearing material in the secondary system
13.	99-2784-8	Revision of UFSAR chapter 15, offsite dose analyses to incorporate the new iodine spiking methodology
14.	99-8530-26	Replacement of ASIC-based cards for 7300 series analog cards
15.	99-10113-18	Use of VIPRE computer code for steady and transient DNBR analysis
16.	00-3229-20	Revision of the analysis for the loss of normal feedwater flow event described in the UFSAR section 15.2.7.
17.	00-3229-27	UFSAR section 15.2.7 update to implement new model E and Delta 94 steam generator loss of normal feedwater analysis
18.	00-6479-20	Add bypass testing capability to RPS and ESFAS
19.	00-8094-36	Incomplete rod insertion evaluation for Unit 1Cycle11
20.	00-8159-2	Revision of UFSAR to reflect changes in the description of a fuel handling accident in the fuel handling building
21.	00-15826-15	Elbow tap flow methodology changes
22.	00-16902-6	Revision to declare one RHR heat exchanger as a spare train
23.	01-9518-6	Revision to allow an increase in pressurizer water level above the program level while in Mode 3
24.	01-10770-2	Revision to reflect the result of RCS over-pressurization analysis in the event of uncontrolled RCCA bank withdrawal at power
25.	01-11964-2	Proposed change to remove AFW valve pit cover in "D" train in both units
26.	01-15007-3	IVC High Energy Line Break Failure Modes and Effects Analysis changes
27.	01-18843-18	Post-LOCA containment hydrogen generation analysis changes
28.	01-18843-20	Irradiation sample analysis method changes
29.	01-18843-21	Addition of a new methodology for evaluating fuel rod design and performance
30.	02-15006-1	Revision to UFSAR as a result of changes made to the spent fuel pool boiling doses calculation
31.	02-19072-34	Special test to determine torsional frequencies on Unit 2 turbine rotors

10CFR50.59 Evaluation Summaries

1. 95-4416-149

Description: This change replaces the existing Beta electronics hardware in the Engineered Safety Features Status Monitoring System with Westinghouse Integrated Computer System controllers which perform the equivalent function.
Summary: The new system meets or exceeds the existing functionality and requirements of the Engineered Safety Features Status Monitoring System.

2. 96-2014-95

Description: This change divides the functionality of the Emergency Response Facility Data Acquisition and Display System (ERFDADS) between the Integrated Computer System and the existing system.
Summary: The new Integrated Computer System meets or exceeds the existing functionality and requirements of the ERFDADS. **A summary of this evaluation should have been included in the 10CFR50.59 summary report sent on January 18, 2001.**

3. 97-0021

Description: This is for a change to the Technical Specification Bases for Low Temperature Overpressure Protection (LTOP). The proposed change corrects the expected pressure in the Reactor Coolant System.
Summary: The existing Basis contains an incorrect RCS pressure value if the Residual Heat Removal System relief valves were actuated to mitigate an LTO condition. The proposed change makes the Technical Specification Basis text description consistent with the design basis calculation. The conditions, bases and safety-related functions of equipment and systems assumed in the Safety Analysis Report are not affected by this change. **A summary of this evaluation should have been included in the 10CFR50.59 summary report sent on December 10, 1998.**

4. 97-0025

Description: The scope of this change is to modify the Main Control Room by replacing the operator consoles with an improved human factors design. This modification will also incorporate portions of the Integrated Computer System (ICS) into the new Control Room consoles to replace the existing Plant Computer System Monitors (Proteus). Additionally, the control room carpeting will be replaced.
Summary: This change has no impact on the Technical Specifications, Offsite Dose Calculation Manual, or Operations Quality Assurance Plan. The change is bounded by the current licensing bases in content (UFSAR), as well as bases of acceptability review (SER).

5. 97-6297-7

Description: This evaluation addresses the UFSAR and physical system changes/impacts resulting from abandoning the 150 ton essential chillers in place.
Summary: A calculation was performed which shows that adequate margin exists for heat removal with a 300 ton chiller per essential chilled water train in affected equipment qualification (EQ) areas under normal, accident and single train safe shutdown conditions.

6. 97-15832-4

Description: This change revises sections in chapter 9 of the UFSAR on water treatment of the Auxiliary Closed Loop System. It is an editorial clarification to enhance the description of the secondary chemical addition process.
Summary: The UFSAR description of water treatment of the Auxiliary Closed Loop System is being consolidated to provide an accurate description of the STP Chemical Treatment Program for the Closed Loop System. The change also removes specific statements about the program and refers to the station chemical addition procedures instead. **A summary of this evaluation should have been included in the 10CFR50.59 summary report sent on January 18, 2001.**

7. 98-0001

Description: This change adds information to the UFSAR describing the addition of a pre-filter and a high efficiency particulate (HEPA) filter to the Reactor Containment Building Supplemental Purge Exhaust system.
Summary: The filter unit is installed in the non-safety related portion of the ductwork. The change also revises the RCB supplemental purge supply system air flow from 5,000 SCFM to 4,000 SCFM to accommodate potentially dirty filter air flow requirements. Radiological consequences described by this change are bounded by those set by 10CFR100 and the bases for Chapter 15 analysis which used 5,000 SCFM. **A summary of this evaluation should have been included in the 10CFR50.59 summary report sent on January 18, 2001.**

8. 98-19444-19

Description: This change is part of the Steam Generator Replacement Project for Unit 2 to incorporate changes associated with the installation of the new steam generators including modifying portions of the Feedwater System and Auxiliary Feedwater System to reroute portions of each system, accommodate new nozzle locations and meet pipe stress criteria.
Summary: The piping will be reinstalled to satisfy existing design requirements in accordance with ASME Sections III and XI. There are no modifications in this design change that would result in a change to the dose mitigating functions of the affected systems and commodities, in any new accident initiators or in any adverse effects on equipment important to safety.

9. 98-19445-6

Description: This change is part of the Steam Generator Replacement Project for Unit 2 to incorporate changes related to the removal and reinstallation of the steam generator blowdown system.
Summary: Revisions to the piping stresses at break locations and highest non-break locations, along with the blowdown rerouting, do not result in any additional pipe break locations. The blowdown piping system still meets all applicable ASME Code stress limits and penetration allowable loads. The probability and consequences of a pipe break in the steam generator blowdown system are unaffected by the revised piping stresses. The probability and consequences of a failure to isolate the steam generator blowdown system, when required to do so, are also unaffected since the revised piping stresses do not impact any of the supporting systems (e.g., valve operation, electrical power to operate valves, instrumentation to signal isolation, and availability of instrument air) required to isolate the system. The consequences of a postulated pipe break at the revised location of the steam generator nozzle for the two-inch blowdown piping have been shown to be acceptable.

10. 99-66-23

Description: This change involves the replacement of the model E steam generators with Delta 94 steam generators.
Summary: The evaluation addresses the UFSAR chapter 15 non-LOCA safety analyses listed in Table 1 including steam generator tube rupture analysis and the subsequent impact on UFSAR Section 4, the TMI Action Plan in the UFSAR Section 7A, and the long-term cooling in UFSAR section 10. The non-LOCA events were re-analyzed/evaluated to support the replacement steam generator program. The evaluation shows that operation of the Delta 94 steam generators is acceptable. **A summary of this evaluation should have been included in the 10CFR50.59 summary report sent on January 18, 2001.**

11. 99-66-64

Description: This change replaces the Model E steam generator UFSAR analyses pertaining to LOCA and MSLB pressure/temperature analyses with the Delta 94 steam generator analyses.
Summary: A review of the principle parameters used in Design Basis High Energy Line Break Accident mass and energy release calculations shows that, in all cases, the Model E steam generator results are bounded by Delta 94 steam generator results. The containment LOCA and MSLB analyses are bounded by the Delta 94 steam generator analyses. The Delta 94 steam generator containment pressure and temperature response analyses have been accepted by the NRC.

12. 99-2157-2

Description: This change revises UFSAR section 5.4.2.1.3 to allow for the possibility of a small amount of copper-bearing material in the secondary system rather than stating that copper has been eliminated.
Summary: There is no change to the NRC-approved secondary water chemistry program. There are no changes to any components or systems. The operating procedures remain the same. There is no change to the testing requirements and acceptance criteria for steam generator tube inspections. **A summary of this evaluation should have been included in the 10CFR50.59 summary report sent on January 18, 2001.**

13. 99-2784-8

Description: This evaluation addresses proposed changes to various sections of the UFSAR resulting from revisions of design basis accident analyses to incorporate the new iodine spiking methodology.
Summary: The following analyses were revised:

- Main Steam Line Break (MSLB) UFSAR Section 15.1.5
- Small Line Break Carrying Primary Coolant Outside Containment (SLBOC) UFSAR Section 15.6.2
- Steam Generator Tube Rupture (SGTR) UFSAR Section 15.6.3

The net result of the changes increased the offsite dose consequence for some of the analyses. However, the offsite doses remain below 10CFR100 and NUREG-0800 limits. Additionally, the increases are considered minimal per criteria used to categorize changes to the UFSAR under 10CFR50.59. The methodology changes discussed in this evaluation have been approved by the NRC.

14. 99-8530-26

Description: This change replaces, on an "as-needed" basis, Westinghouse supplied 7300 process instrumentation cards with new cards developed by Westinghouse using Application Specific Integrated Circuit (ASIC) technology.
Summary: The ASIC-Based Replacement Module (ABRM) is intended to be a direct card-for-card replacement module for the Westinghouse 7300 process instrumentation. The ABRM will be used as replacement for the 7300 card on an "as-fail basis" or as a targeted replacement for specific types of card showing an increased failure rate. Westinghouse has performed a substantial portion of this evaluation on a generic basis and the NRC has accepted this evaluation in its Safety Evaluation Report. As directed by the NRC SER, the results of this generic acceptance are to be supplemented by a site-specific evaluation of those issues that could not be addressed generically. This evaluation found that application of ABRM's in the Westinghouse 7300 system is appropriately bounded by the existing STP design bases and associated analyses and is therefore technically acceptable for use.

15. 99-10113-18

Description: The subject change revises the UFSAR to allow the use of the VIPRE computer code for the evaluation of core thermal hydraulic parameters, including the departure from nucleate boiling ratio (DNBR) analysis.
Summary: The VIPRE code will be used to perform licensing basis analyses and evaluations for Robust Fuel Assemblies. The NRC in WCAP-14565-P-A approved the use of the VIPRE computer code for the evaluation of core thermal hydraulic parameters, including Departure from Nucleate Boiling Ratio (DNBR) analysis. The computer code will be used in accordance with the restrictions identified in the NRC Safety Evaluation Report.

16. 00-3229-20

Description: The proposed change revises the analysis for the loss of normal feedwater flow event described in UFSAR Section 15.2.7.
Summary: The revised analysis takes credit for operator actions not originally credited for this event and the availability of four steam generators. The results of the analysis show that all acceptance limits are satisfied. The new operator actions have been evaluated using criteria in IN-97-78. The operator actions have been approved by the NRC and do not need approval before inclusion into the design basis for this event. The LCO for AFW Pump "A" in Technical specification 3.7.1.2 will be administratively controlled to comply with the LCO for the other AFW pumps.

17. 00-3229-27

Description: This evaluation documents the revised Model E and Delta 94 steam generator loss of normal feedwater analyses.
Summary: The new analyses were performed for 0% tube plugging, following the determination that the assumption of minimum tube plugging, instead of maximum tube plugging, was conservative for the margin-to-pressurizer-overfill analysis. The revised Model E analysis takes credit for steam generator PORV operation and operator action at 30 minutes to start a third AFW pump. The revised Delta 94 steam generator analysis credits operator action at 15 minutes to start a third AFW pump. The Delta 94 analysis also includes the modeling of pressurizer backup heated operation on high pressurizer level deviation. The reduction in the operator response time requirement from 30 minutes to 15 minutes in the Delta 94 analysis is supported by the validation package for the applicable emergency operating procedure. The revised analyses meet all applicable acceptance criteria. The new analyses were performed using the approved methodologies.

18. 00-6479-20

Description: The purpose of this modification is to change the Reactor Protection System (RPS) and Engineered Safety Features Actuation System (ESFAS) logic to add the capability of placing channels into a "bypass" mode in addition to the existing "trip" mode. The activity reviewed was for equipment changes only; a separate Technical Specification change was obtained to permit relaxation of the allowed outage times and bypass test times.
Summary: This modification will reduce the potential for spurious turbine and reactor trips as well as ESFAS actuations, thereby increasing plant reliability. With the addition of the bypass equipment, spurious reactor trip or safeguards actuation will be avoided, increasing plant availability. The reduction in the number of reactor trips and safeguard actuation will reduce challenges to the RPS and avoid transients associated with reactor trips and safeguard actuations. Based on a comprehensive review of the bypass equipment system and the design and testing documentation provided in support of compliance with the regulatory requirements listed in the evaluation, it has been determined that the applicable Instrumentation and Control system licensing criteria have been satisfied.

19. 00-8094-36

Description: This evaluation shows that the South Texas Project has taken reasonable precautions to ensure that failure of the RCCAs to fully insert will not occur, or otherwise be limited, for the duration of Unit 1 Cycle 11.
Summary: The Safety Analysis provides bounding results with respect to the Reload Safety Analysis Checklist if RCCAs fail to fully insert. The South Texas Project has taken actions to both minimize the possibility of the Incomplete Rod Cluster Control Assembly Insertion (IRI) phenomena occurring in Unit 1 Cycle 11 (U1 C11), and to ensure that adequate shutdown margin and trip reactivity exist if IRI does occur. The subject condition of the RCCAs failing to insert, given the bounding scenarios examined, is acceptable.

20. 00-8159-2

Description: This evaluation considers revision of UFSAR section 15.7.4.2, and Tables 15.7-9 and -10, to reflect changes in the description of a Fuel Handling Accident (FHA) in the Fuel Handling Building (FHB).
Summary: Changes that were incorporated include revising Table 15.7-9 to increase the radial peaking factor assumed for the damaged assembly from 1.65 to 1.70, changing the closure time for the FHB Exhaust isolation damper in Table 15.7-9 from 20 seconds to 42 seconds, and modifying the offsite doses in Table 15.7-10 to present the results of the revision of the FHA in the FHB that utilizes the changes listed above. The increase in consequences for the FHA in the FHB from the parameter changes is less than ten percent (10%) of the differential between the current value present in the UFSAR and the 10CFR100 regulatory limit and remains below the applicable Standard Review Plan guideline for the FHA.

21. 00-15826-15

Description: The purpose of this modification is to change the methodology used to calculate the Reactor Coolant System (RCS) flow rate measurement uncertainty using the elbow tap method to WCAP 15287, reduce the Reactor Coolant System flow rate measurement uncertainty from 2.8% to 2.1% using the elbow tap methodology, and retain the 2.8% Reactor Coolant System flow rate measurement uncertainty using the precision flow calorimetric method.

Summary: This change in methodology results in an increase to the Baseline Calorimetric Flow that is used to calculate RCS flow using the elbow tap methodology. The methodology used in WCAP 15287 is the same as approved by the NRC in WCAP 15404 for Seabrook. In addition WCAP 15287 performs the cold leg elbow tap RCS flow uncertainty with new Rosemount 1154HP6RB transmitters. The new Rosemount transmitters result in a decrease in the RCS flow measurement uncertainty from 2.6% to 2.1%. These changes will permit the use of either the precision calorimetric RCS flow measurement method with an uncertainty of 2.8% or the elbow tap RCS flow measurement method with an uncertainty of 2.1%. This change is an operational enhancement to provide additional RCS flow margin.

22. 00-16902-6

Description: This change revises the UFSAR to declare one RHR heat exchanger as a spare train as defined in NEI 99-02, "Regulatory Assessment Performance Indicator Guide," Rev. D, Nuclear Energy Institute.

Summary: This change does not propose to change the Operability of the RHR heat exchanger but only pertains to the cooling medium. The heat exchangers will remain intact and able to provide LHSI and RHR flows to the heat exchanger. No physical change to the plant will be made. Results of LOCA containment pressure and temperature analyses with one RHR train in operation demonstrates that all acceptance limits are satisfied. The maximum containment pressure remains at or below 41.2 psig, which is less than the design pressure of 56.5 psig. The LOCA containment pressure and temperature profiles remain within the equipment qualification design criteria limits. The post-LOCA containment hydrogen concentration remains below design limits. There is no adverse impact on containment isolated pipe pressurization analysis, and sufficient margin to ASME Code allowable pressure continues to exist.

23. 01-9518-6

Description: The proposed change allows increasing the pressurizer level up to 55% in Mode 3.

Summary: An analysis of the Chemical and Volume Control Systems Malfunction That Increases Reactor Coolant Inventory accident described in UFSAR Section 15.5.2 was performed in Mode 3. The results of the analysis show that the operator has greater than 10 minutes to correct the potential overfill condition. The operator response time is consistent with the current response time for accident. The results of the analysis show that all acceptance limits are satisfied.

24. 01-10770-2

Description: The proposed change revises various UFSAR sections to reflect the analysis for over-pressurization of the reactor coolant system in the uncontrolled RCCA bank-withdrawal-at-power event.

Summary: In the analysis of the uncontrolled RCCA bank-withdrawal-at-power event, the protection against reactor coolant system over-pressurization is credited to both of the high pressurizer pressure reactor trip and the power range positive neutron flux rate reactor trip functions. The power range positive neutron flux rate reactor trip function is not reflected in the list of trip points and time delays used in the accident analysis table (Table 15.0-4) and instrument response time table (Table 16.1-3) of the UFSAR. The UFSAR change revises the licensing basis to reflect this. Technical Specification Basis 2.2.1 is also revised to incorporate this fact. The proposed change is considered a change to methodology described in the UFSAR. The analysis of the Uncontrolled RCCA Bank Withdrawal at Power event identified that the power range neutron flux high rate reactor trip is used to ensure the RCS pressure stays below the acceptance limit. There is no change to the analytical result described in the UFSAR.

25. 01-11964-2

Description: The proposed change is a compensatory action to remove the auxiliary feedwater valve pit cover to the auxiliary feedwater train allowing local operator action to secure the specified auxiliary feedwater valve within one hour of a feedwater break event.

Summary: The proposed change is a compensatory action which meets the licensing limits for missile protection and is in compliance with the security plan. Security will provide adequate compensatory action in compliance with the physical security plan while the auxiliary feedwater valve pit cover is removed. The long term cooling analysis was reviewed and it was determined that a time of one hour for closure of the specified valve was adequate.

26. 01-15007-3

Description: This evaluation documents the results of the HELB FMEA analyses in UFSAR Sections 10.3, 10.4.7, 10.4.8 and 10.4.9 and the associated Tables.

Summary: The evaluation considered only the IVC electrical equipment/components needed to mitigate the consequences of an accident in the IVC. The evaluation identified electrical equipment in the IVC required for mitigating the consequences of an accident that must be qualified for "harsh" environmental conditions. All other electrical equipment is downgraded to "mild" environment conditions. No evaluation methodology was changed, and no equipment or component needs to be requalified.

27. 01-18843-18

Description: The proposed change revises UFSAR Sections 6.2.5 and associated Tables and Figures to incorporate results of the post-LOCA containment hydrogen generation analysis.
Summary: The analysis determined the maximum hydrogen concentration in the containment following a design-basis LOCA. Two major changes were made to the current licensing basis analysis:

- Revised UFSAR 6.2.5 to clarify that the hydrogen concentration limit is not 3.5 v/o but 4.0 v/o per RG 1.7.
- Changed evaluation methodology from the current Bechtel methodology to one used by Westinghouse.

The proposed change uses a new method of analysis not previously described in the UFSAR. However, the methodology is the same as the methodology previously accepted by NRC for plants similar to STP such as Comanche Peak. Therefore, this is acceptable for use at STP. The results of the analysis show that the 4.0 volume percent hydrogen limit, specified in Regulatory Guide 1.7, will not be exceeded even if only one of the two hydrogen recombiners is placed in service per STP Emergency Operating Procedures. No new Operator Actions are required. The results of the analysis are applicable to STP with either Model E or Delta 94 steam generators, and include the 1.4% power uprate.

28. 01-18843-20

Description: The activity evaluated by this report is the replacement of the methodology used in the STP UFSAR to evaluate reactor pressure vessel fast neutron fluence with a method approved by the NRC in Regulatory Guide (RG) 1.190.
Summary: The activity is part of the adoption of a 1.4% increase in reactor power. The methods described in RG 1.190 are used to assess the pressure vessel lifetime exposure projections based on the vessel exposure history inferred from irradiation samples, including the effect of higher power levels on vessel fluence. This change is purely analytical in nature. The change implements a methodology approved by the NRC in RG 1.190. As a change in analytical methods, this activity creates no new accident initiators, no changes in accident frequency, and no increase in any accident consequences. The results of the analysis show that no fission product barrier design limits are exceeded. This change implements a method approved by the NRC for its intended purpose.

29. 01-18843-21

Description: The activity evaluated by this report is the addition of new methodology referenced in the STP UFSAR to evaluate fuel rod design and performance.
Summary: The activity is part of the adoption of a 1.4% increase in reactor power. The method incorporated by reference is the NRC-approved Westinghouse topical report WCAP-15063-P-A. This topical describes advances in Westinghouse fuel design and performance evaluation methods used to ensure that fuel rods are maintained within design bases. This change is an addition to, rather than a replacement of, the methods already referenced in the UFSAR. This change is purely analytical in nature. As a change in analytical methods, this activity creates no new accident initiators, no changes in accident frequency, and no increase in any accident consequences. The results of the analysis show that no fission product barrier design limits are exceeded.

30. 02-15006-1

Description: This evaluation addresses proposed changes to Chapter 9.1 of the UFSAR resulting from the revision of NC-6056, "Spent Fuel Pool Boiling Doses".
Summary: The revision of NC-6056 includes using ICRP-30 Dose Conversion Factors in lieu of ICRP-2, and commencing core offload at 42 hours after shutdown. The net result of the changes was an increase in the offsite dose consequence for the analysis; however, the offsite doses remain below 10CFR100 limits. The increases are considered minimal per 10CFR50.59 criteria. Also, the methodology changes discussed in this evaluation have been approved by the NRC.

31. 02-19072-34

Description: The proposed change is a special test needed to determine the torsional natural frequencies for the turbine rotors on Unit 2 due to excessive turbine vibrations.

Summary: The Unit 2 Main Turbine/Generator Rotor Torsional Response Test will be run with the unit at 12% to 14% reactor power. A temporary modification will place the main generator and main transformers in an abnormal condition; however, the test will control the output current of the generator to low levels by under-exciting the generator. Protective relaying for the main generator that will be defeated by the temporary modification will not be necessary because the generator will not be connected to the grid. Offsite power requirements during the test will still be met. There is no impact on SSCs important to safety and/or credited in accident analyses. There is no increase in probability of any Condition II, III or IV transients. Any sudden load decrease or load increase transients are bounded by existing Condition I analyses. All existing licensing-basis accident analyses remain applicable and bounding.