Mr. James M. Levine
Senior Vice President, N   ear
Arizona Public Service Company
P. O. Box 53999
Phoenix, AZ 85072-3999

August 5, 1999

SUBJECT: PALO VERDE NUCLEAR GENERATING STATION UNITS 1, 2, AND 3 - ISSUANCE OF AMENDMENTS RE: STEAM GENERATOR TUBE SLEEVING (TAC NOS. M98920, M98921 AND M98922)

Dear Mr. Levine:

The Commission has issued the enclosed Amendment No. 120 to Facility Operating License Nos. NPF-41, NPF-51, and NPF-74 for the Palo Verde Nuclear Generating Station, Units 1, 2, and 3, respectively. The amendments consist of changes to the Technical Specifications (TSs) in response to your application dated May 23, 1997, as supplemented by letters dated September 27, 1998, and May 26, 1999.

The amendments will allow the installation of ABB Combustion Engineering (ABB-CE) leak-tight sleeves in defective steam generator tubes as a tube repair method. To facilitate NRC approval of your TS amendment request, the TS changes proposed for the Palo Verde units contained in your May 26, 1999, letter are similar to the TS changes approved previously for other nuclear power plants. The staff is working with the nuclear power industry to develop performance-based TSs that will allow for more flexibility. Once this process is complete and appropriate TS requirements finalized, the NRC will be in a position to approve TS amendment requests based on this approach.

A copy of the related Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's next biweekly Federal Register notice.

Sincerely,
/s/
N. Kalyanam, Project Manager, Section 2
Project Directorate IV & Decommissioning
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. STN 50-528, STN 50-529, and STN 50-530

Enclosures:
1. Amendment No. 120 to NPF-41
2. Amendment No. 120 to NPF-51
3. Amendment No. 120 to NPF-74
4. Safety Evaluation

cc w/encls: See next page

DFO1

CP-1

DISTRIBUTION

Docket File	WBeckner	SRichards (clo)	OGC	GHill(6)
PUBLIC	PHarrell, RIV	RScholl (email SE)	ACRS	
PDIV-2 Rdg	LHurley, RIV	JTsao	JKilcrease, RIV	*No major changes to SE

To receive a copy of this document, indicate "C" in the box

OFFICE	PDIV-2/PM	C	PDIV-D/LA	C	EMCB/BC	OGC NLO with comment	PDIV-2/SC
NAME	NKalyanam		CJamerson		WBateman	RWeisman	SDembek
DATE	7/29/99		8/4/99		08/3/99	08/04/99	8/5/99

DOCUMENT NAME: G:\PDIV-2\PaloVerde\Amd98920.wpd

120025

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

August 5, 1999

Mr. James M. Levine
Senior Vice President, Nuclear
Arizona Public Service Company
P. O. Box 53999
Phoenix, AZ 85072-3999

SUBJECT: PALO VERDE NUCLEAR GENERATING STATION UNITS 1, 2, AND 3 - ISSUANCE OF AMENDMENTS RE: STEAM GENERATOR TUBE SLEEVING (TAC NOS. M98920, M98921 AND M98922)

Dear Mr. Levine:

The Commission has issued the enclosed Amendment No. 120 to Facility Operating License Nos. NPF-41, NPF-51, and NPF-74 for the Palo Verde Nuclear Generating Station, Units 1, 2, and 3, respectively. The amendments consist of changes to the Technical Specifications (TSs) in response to your application dated May 23, 1997, as supplemented by letters dated September 27, 1998, and May 26, 1999.

The amendments will allow the installation of ABB Combustion Engineering (ABB-CE) leak-tight sleeves in defective steam generator tubes as a tube repair method. To facilitate NRC approval of your TS amendment request, the TS changes proposed for the Palo Verde units contained in your May 26, 1999, letter are similar to the TS changes approved previously for other nuclear power plants. The staff is working with the nuclear power industry to develop performance-based TSs that will allow for more flexibility. Once this process is complete and appropriate TS requirements finalized, the NRC will be in a position to approve TS amendment requests based on this approach.

A copy of the related Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's next biweekly Federal Register notice.

Sincerely,

N. Kalyanam, Project Manager, Section 2
Project Directorate IV & Decommissioning
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. STN 50-528, STN 50-529,
and STN 50-530

Enclosures:
1. Amendment No. 120 to NPF-41
2. Amendment No. 120 to NPF-51
3. Amendment No. 120 to NPF-74
4. Safety Evaluation

cc w/encls: See next page

cc:

Mr. Steve Olea
Arizona Corporation Commission
1200 W. Washington Street
Phoenix, AZ 85007

Douglas Kent Porter
Senior Counsel
Southern California Edison Company
Law Department, Generation Resources
P.O. Box 800
Rosemead, CA 91770

Senior Resident Inspector
U.S. Nuclear Regulatory Commission
P. O. Box 40
Buckeye, AZ 85326

Regional Administrator, Region IV
U.S. Nuclear Regulatory Commission
Harris Tower & Pavillion
611 Ryan Plaza Drive, Suite 400
Arlington, TX 76011-8064

Chairman, Board of Supervisors
ATTN: Chairman
301 W. Jefferson, 10th Floor
Phoenix, AZ 85003

Mr. Aubrey V. Godwin, Director
Arizona Radiation Regulatory Agency
4814 South 40 Street
Phoenix, AZ 85040

Ms. Angela K. Krainik, Manager
Nuclear Licensing
Arizona Public Service Company
P.O. Box 52034
Phoenix, AZ 85072-2034

Mr. John C. Horne
Vice President, Power Generation
El Paso Electric Company
2702 N. Third Street, Suite 3040
Phoenix, AZ 85004

Mr. David Summers
Public Service Company of New Mexico
414 Silver SW, #1206
Albuquerque, NM 87102

Mr. Jarlath Curran
Southern California Edison Company
5000 Pacific Coast Hwy Bldg DIN
San Clemente, CA 92672

Mr. Robert Henry
Salt River Project
6504 East Thomas Road
Scottsdale, AZ 85251

Terry Bassham, Esq.
General Counsel
El Paso Electric Company
123 W. Mills
El Paso, TX 79901

Mr. John Schumann
Los Angeles Department of Water & Power
Southern California Public Power Authority
P.O. Box 51111, Room 1255-C
Los Angeles, CA 90051

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

ARIZONA PUBLIC SERVICE COMPANY, ET AL.

DOCKET NO. STN 50-528

PALO VERDE NUCLEAR GENERATING STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 120
License No. NPF-41

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by the Arizona Public Service Company (APS or the licensee) on behalf of itself and the Salt River Project Agricultural Improvement and Power District, El Paso Electric Company, Southern California Edison Company, Public Service Company of New Mexico, Los Angeles Department of Water and Power, and Southern California Public Power Authority dated May 23, 1997, as supplemented September 27, 1998, and May 26, 1999, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C(2) of Facility Operating License No. NPF-41 is hereby amended to read as follows:

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 120, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated into this license. APS shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan, except where otherwise stated in specific license conditions.

3. This license amendment is effective as of the date of issuance, and shall be implemented within 45 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Stephen Dembek, Chief, Section 2
Project Directorate IV & Decommissioning
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: August 5, 1999

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

ARIZONA PUBLIC SERVICE COMPANY, ET AL.

DOCKET NO. STN 50-529

PALO VERDE NUCLEAR GENERATING STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 120
License No. NPF-51

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by the Arizona Public Service Company (APS or the licensee) on behalf of itself and the Salt River Project Agricultural Improvement and Power District, El Paso Electric Company, Southern California Edison Company, Public Service Company of New Mexico, Los Angeles Department of Water and Power, and Southern California Public Power Authority dated May 23, 1997, as supplemented September 27, 1998, and May 26, 1999, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C(2) of Facility Operating License No. NPF-51 is hereby amended to read as follows:

(2) <u>Technical Specifications and Environmental Protection Plan</u>

The Technical Specifications contained in Appendix A, as revised through Amendment No. 120, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated into this license. APS shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan, except where otherwise stated in specific license conditions.

3. This license amendment is effective as of the date of issuance, and shall be implemented within 45 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Stephen Dembek

Stephen Dembek, Chief, Section 2
Project Directorate IV & Decommissioning
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: August 5, 1999

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

ARIZONA PUBLIC SERVICE COMPANY, ET AL.

DOCKET NO. STN 50-530

PALO VERDE NUCLEAR GENERATING STATION, UNIT 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No.120
License No. NPF-74

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by the Arizona Public Service Company (APS or the licensee) on behalf of itself and the Salt River Project Agricultural Improvement and Power District, El Paso Electric Company, Southern California Edison Company, Public Service Company of New Mexico, Los Angeles Department of Water and Power, and Southern California Public Power Authority dated May 23, 1997, as supplemented September 27, 1998, and May 26, 1999, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C(2) of Facility Operating License No. NPF-74 is hereby amended to read as follows:

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 120, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated into this license. APS shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan, except where otherwise stated in specific license conditions.

3. This license amendment is effective as of the date of issuance, and shall be implemented within 45 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Stephen Dembek, Chief, Section 2
Project Directorate IV & Decommissioning
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: August 5, 1999

ATTACHMENT TO LICENSE AMENDMENT NO. 120

FACILITY OPERATING LICENSE NOS. NPF-41, NPF-51, AND NPF-74

DOCKET NOS. STN 50-528, STN 50-529, AND STN 50-530

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

REMOVE	INSERT
3.4.14-1	3.4.14-1
5.0-1	5.1-1
5.0-2	5.2-1
5.0-3	5.2-2
5.0-4	5.2-3
5.0-5	5.3-1
5.0-6	5.4-1
5.0-7	5.5-1
5.0-8	5.5-2
5.0-9	5.5-3
5.0-10	5.5-4
5.0-11	5.5-5
5.0-12	5.5-6
5.0-13	5.5-7
5.0-14	5.5-8
5.0-15	5.5-9
5.0-16	5.5-10
5.0-17	5.5-11
5.0-18	5.5-12
5.0-19	5.5-13
5.0-20	5.5-14
5.0-21	5.5-15
5.0-22	5.5-16
5.0-23	5.5-17
5.0-24	5.5-18
5.0-25	5.5-19
5.0-26	5.5-20
5.0-27	5.5-21
5.0-28	5.5-22
5.0-29	5.5-23
5.0-30	5.5-24
5.0-31	5.6-1
5.0-32	5.6-2
5.0-33	5.6-3
--	5.6-4
--	5.6-5
--	5.6-6
--	5.7-1
--	5.7-2

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.14 RCS Operational LEAKAGE

LCO 3.4.14 RCS operational LEAKAGE shall be limited to:

a. No pressure boundary LEAKAGE;

b. 1 gpm unidentified LEAKAGE;

c. 10 gpm identified LEAKAGE; and

d. 150 gallons per day primary to secondary LEAKAGE through any one SG.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. RCS LEAKAGE not within limits for reasons other than pressure boundary LEAKAGE.	A.1 Reduce LEAKAGE to within limits.	4 hours
B. Required Action and associated Completion Time of Condition A not met. OR Pressure boundary LEAKAGE exists.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours
C. One or more SGs inoperable.	C.1 Enter LCO 3.0.3.	Immediately

5.0 ADMINISTRATIVE CONTROLS

5.1 Responsibility

5.1.1 The Department Leader, Operations shall be responsible for overall unit operation and shall delegate in writing the succession to this responsibility during his absence.

The Department Leader, Operations or his designee shall approve, prior to implementation, each proposed test, experiment or modification to systems or equipment that affect nuclear safety.

5.1.2 The Control Room Supervisor (CRS) shall be responsible for the control room command function. During any absence of the CRS from the control room while the unit is in MODE 1, 2, 3, or 4, an individual with an active Senior Reactor Operator (SRO) license shall be designated to assume the control room command function. During any absence of the CRS from the control room while the unit is in MODE 5 or 6, an individual with an active SRO license or Reactor Operator license shall be designated to assume the control room command function.

5.0 ADMINISTRATIVE CONTROLS

5.2 Organization

5.2.1 Onsite and Offsite Organizations

Onsite and offsite organizations shall be established for unit operation and corporate management, respectively. The onsite and offsite organizations shall include the positions for activities affecting safety of the nuclear power plant.

a. Lines of authority, responsibility, and communication shall be defined and established throughout highest management levels, intermediate levels, and all operating organization positions. These relationships shall be documented and updated, as appropriate, in organization charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These requirements shall be documented in the UFSAR;

b. The Vice President, Nuclear Production shall be responsible for overall safe operation of the plant and shall have control over those onsite activities necessary for safe operation and maintenance of the plant;

c. The Senior Vice President, Nuclear shall have corporate responsibility for overall plant nuclear safety and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the plant to ensure nuclear safety; and

d. The individuals who train the operating staff, carry out health physics, or perform quality assurance functions may report to the appropriate onsite manager; however, these individuals shall have sufficient organizational freedom to ensure their independence from operating pressures.

5.2.2 Unit Staff

The unit staff organization shall include the following:

a. A non-licensed operator shall be assigned to each reactor containing fuel and an additional non-licensed operator

(continued)

5.2 Organization

5.2.2 Unit Staff (continued)

shall be assigned for each control room from which a reactor is operating in MODES 1, 2, 3, or 4.

b. Shift crew composition shall meet the requirements stipulated herein and in 10 CFR 50.54(m). Shift crew composition may be less than the minimum requirement of 10 CFR 50.54(m)(2)(i) and 5.2.2.a and 5.2.2.f for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements.

c. A Radiation Protection Technician shall be on site when fuel is in the reactor. The position may be vacant for not more than 2 hours, in order to provide for unexpected absence, provided immediate action is taken to fill the required position.

d. Administrative procedures shall be developed and implemented to limit the working hours of unit staff who perform safety related functions (e.g., licensed SROs, licensed ROs, radiation protection technicians, auxiliary operators, and key maintenance personnel).

The controls shall include guidelines on working hours that ensure adequate shift coverage shall be maintained without routine heavy use of overtime.

Any deviation from the working hour guidelines shall be authorized in advance by personnel at the Director level or designees, in accordance with approved administrative procedures and with documentation of the basis for granting the deviation.

Controls shall be included in the procedures such that individual overtime shall be reviewed monthly by these authorized individuals or designees to ensure that excessive hours have not been assigned. Routine deviation from the above guidelines is not authorized.

(continued)

5.2 Organization

5.2.2 Unit Staff (continued)

e. The Operations Department Leader or Operations Supervisor shall hold an SRO license.

f. The Shift Technical Advisor (STA) shall provide advisory technical support to the Shift Manager in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit. In addition, the STA shall meet the qualifications specified by the Commission Policy Statement on Engineering Expertise on Shift.

5.0 ADMINISTRATIVE CONTROLS

5.3 Unit Staff Qualifications

5.3.1 Each member of the unit staff shall meet or exceed the minimum qualifications of Regulatory Guide 1.8, September 1975 and ANSI/ANS 3.1-1978, except the Director, Site Radiation Protection shall meet or exceed the qualification of Regulatory Guide 1.8, September 1975, and the Shift Technical Advisor shall have a bachelor's degree or equivalent in a scientific or engineering discipline with specific training in plant design and plant operating characteristics, including transients and accidents.

5.3.2 For the purpose of 10 CFR 55.4, a licensed senior reactor operation (SRO) and a licensed reactor operator (RO) are those individuals who, in addition to meeting the requirements of TS 5.3.1, perform the functions described in 10 CFR 50.54(m).

5.0 ADMINISTRATIVE CONTROLS

5.4 Procedures

5.4.1 Written procedures shall be established, implemented, and maintained covering the following activities:

a. The applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978;

b. The emergency operating procedures required to implement the requirements of NUREG-0737 and to NUREG-0737, Supplement 1, as stated in Generic Letter 82-33;

c. Quality assurance for effluent and environmental monitoring;

d. Fire Protection Program implementation; and

e. All programs specified in Specification 5.5.

f. Modification of core protection calculator (CPC) addressable constants. These procedures shall include provisions to ensure that sufficient margin is maintained in CPC type I addressable constants to avoid excessive operator interaction with CPCs during reactor operation.

Modifications to the CPC software (including changes of algorithms and fuel cycle specific data) shall be performed in accordance with the most recent version of "CPC Protection Algorithm Software Change Procedure," CEN-39(A)-P, which has been determined to be applicable to the facility. Additions or deletions to CPC addressable constants or changes to addressable constant software limit values shall not be implemented without prior NRC approval.

5.0 ADMINISTRATIVE CONTROLS

5.5 Programs and Manuals

The following programs shall be established, implemented, and maintained.

5.5.1 Offsite Dose Calculation Manual (ODCM)

a. The ODCM shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm and trip setpoints, and in the conduct of the radiological environmental monitoring program; and

b. The ODCM shall also contain the radioactive effluent controls and radiological environmental monitoring activities and descriptions of the information that should be included in the Annual Radiological Environmental Operating, and Radioactive Effluent Release Reports required by Specification 5.6.2 and Specification 5.6.3.

Licensee initiated changes to the ODCM:

a. Shall be documented and records of reviews performed shall be retained. This documentation shall contain:

   1. Sufficient information to support the change(s) together with the appropriate analyses or evaluations justifying the change(s),

   2. A determination that the change(s) maintain the levels of radioactive effluent control required by 10 CFR 20.1302, 40 CFR 190, 10 CFR 50.36a, and 10 CFR 50, Appendix I, and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations;

b. Shall become effective after the approval of the Director, Site Chemistry; and

c. Shall be submitted to the NRC in the form of a complete, legible copy of the entire ODCM as a part of or concurrent with the Radioactive Effluent Release Report for the period of the report in which any change in the ODCM was made. Each change shall be identified by markings in the margin of

(continued)

5.5 Programs and Manuals

5.5.1 Offsite Dose Calculation Manual (ODCM) (continued)

the affected pages, clearly indicating the area of the page that was changed, and shall indicate the date (i.e., month and year) the change was implemented.

5.5.2 Primary Coolant Sources Outside Containment

This program provides controls to minimize leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to levels as low as practicable. The systems include recirculation portion of the high pressure injection system, the shutdown cooling portion of the low pressure safety injection system, the post-accident sampling subsystem of the reactor coolant sampling system, the containment spray system, the post-accident sampling return piping of the radioactive waste gas system, the post-accident sampling return piping of the liquid radwaste system, and the post-accident containment atmosphere sampling piping of the hydrogen monitoring subsystem. The program shall include the following:

a. Preventive maintenance and periodic visual inspection requirements; and

b. Integrated leak test requirements for each system at refueling cycle intervals or less.

5.5.3 Post Accident Sampling

This program provides controls that ensure the capability to obtain and analyze reactor coolant, radioactive gases, and particulates in plant gaseous effluents and containment atmosphere samples under accident conditions. The program shall include the following:

a. Training of personnel;

b. Procedures for sampling and analysis; and

c. Provisions for maintenance of sampling and analysis equipment.

(continued)

5.5 Programs and Manuals

5.5.4 Radioactive Effluent Controls Program

This program conforms to 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to members of the public from radioactive effluents as low as reasonably achievable. The program shall be contained in the ODCM, shall be implemented by procedures, and shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

a. Limitations on the functional capability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM;

b. Limitations on the concentrations of radioactive material released in liquid effluents to unrestricted areas, conforming to 10 times the concentration values in Appendix B, Table 2, Column 2 to 10 CFR 20.1001-20.2402;

c. Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10 CFR 20.1302 and with the methodology and parameters in the ODCM;

d. Limitations on the annual and quarterly doses or dose commitment to a member of the public from radioactive materials in liquid effluents released from each unit to unrestricted areas, conforming to 10 CFR 50, Appendix I;

e. Determination of cumulative and projected dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days;

f. Limitations on the functional capability and use of the liquid and gaseous effluent treatment systems to ensure that appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a period of 31 days would exceed 2% of the guidelines for the annual dose or dose commitment, conforming to 10 CFR 50, Appendix I;

(continued)

5.5 Programs and Manuals

5.5.4 Radioactive Effluent Controls Program (continued)

g. Limitations on the dose rate resulting from radioactive material released in gaseous effluents from the site to areas at or beyond the site boundary:

1. For noble gases: less than or equal to a dose rate of 500 mrems/yr to the total body and less than or equal to a dose rate of 3000 mrems/yr to the skin, and

2. For iodine-131, iodine-133, tritium, and for all radionuclides in particulate form with half-lives greater than 8 days: less than or equal to a dose rate of 1500 mrems/yr to any organ;

h. Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I;

i. Limitations on the annual and quarterly doses to a member of the public from iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half lives > 8 days in gaseous effluents released from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I; and

j. Limitations on the annual dose or dose commitment to any member of the public beyond the site boundary due to releases of radioactivity and to radiation from uranium fuel cycle sources, conforming to 40 CFR 190.

5.5.5 Component Cyclic or Transient Limit

This program provides controls to track the UFSAR Section 3.9.1.1 cyclic and transient occurrences to ensure that components are maintained within the design limits.

5.5.6 Pre-Stressed Concrete Containment Tendon Surveillance Program

This program provides controls for monitoring any tendon degradation in pre-stressed concrete containments, including effectiveness of its corrosion protection medium, to ensure containment structural integrity. The program shall include baseline measurements prior to initial operations. The Tendon Surveillance Program, inspection frequencies, and acceptance criteria shall be in accordance with Regulatory Guide 1.35, as described in Section 1.8 of the UFSAR.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Tendon Surveillance Program inspection frequencies.

(continued)

5.5 Programs and Manuals

5.5.7 Reactor Coolant Pump Flywheel Inspection Program

This program shall provide for the inspection of each reactor coolant pump flywheel per the recommendations of regulatory position c.4.b of Regulatory Guide 1.14, Revision 0, October 1971.

5.5.8 Inservice Testing Program

This program provides controls for inservice testing of ASME Code Class 1, 2, and 3 components including applicable supports. The program shall include the following:

a. Testing frequencies specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as follows:

ASME Boiler and Pressure Vessel Code and applicable Addenda terminology for inservice testing activities	Required Frequencies for performing inservice testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Every 9 months	At least once per 276 days
Yearly or annually	At least once per 366 days
Biennially or every 2 years	At least once per 731 days

b. The provisions of SR 3.0.2 are applicable to the above required Frequencies for performing inservice testing activities;

c. The provisions of SR 3.0.3 are applicable to inservice testing activities; and

d. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any TS.

(continued)

5.5 Programs and Manuals (continued)

5.5.9 Steam Generator (SG) Tube Surveillance Program

This program provides controls for the Inservice Inspection of steam generator tubes and tube sleeves to ensure that structural integrity of this portion of the RCS is maintained. The program shall include the following:

5.5.9.1 Steam Generator Sample Selection and Inspection - Each steam generator shall be determined OPERABLE during shutdown by selecting and inspecting tubes and tube sleeves in at least the minimum number of steam generators specified in Table 5.5.9-1.

5.5.9.2a Steam Generator Tube Sample Selection and Inspection - The steam generator tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Table 5.5.9-2. The inservice inspection of steam generator tubes shall be performed at the frequencies specified in 5.5.9.3 and the inspected tubes shall be verified acceptable per the acceptance criteria of 5.5.9.4. The tubes selected for each inservice inspection shall include at least 3% of the total number of tubes in all steam generators; the tubes selected for these inspections shall be selected on a random basis except:

a. Where experience in similar plants with similar water chemistry indicates critical areas to be inspected, then at least 50% of the tubes inspected shall be from these critical areas.

b. The first sample of tubes selected for each inservice inspection (subsequent to the preservice inspection) of each steam generator shall include:

1. All nonplugged tubes that previously had detectable wall penetrations (greater than 20% and not sleeved in that area).

2. Tubes in those areas where experience has indicated potential problems.

3. A tube inspection (pursuant to Specification 5.5.9.4.a.10.) shall be performed on each selected tube. If any selected tube does not permit the passage of the eddy current probe for a tube inspection, this shall be recorded and an adjacent tube shall be selected and subjected to a tube inspection.

c. The tubes selected as the second and third samples (if required by Table 5.5.9-2) during each inservice inspection may be subjected to a partial tube inspection provided:

(continued)

5.5 Programs and Manuals (continued)

5.5.9.2a Steam Generator Tube Sample Selection and Inspection (continued)

1. The tubes selected for these samples include the tubes from those areas of the tube sheet array where tubes with imperfections were previously found.

2. The inspections include those portions of the tubes where imperfections were previously found.

The results of each inspection sample shall be classified into one of the following three categories (this classification shall apply to the inspection of tubes and treated exclusive of the sleeve inspections in 5.5.9.2b):

Category	Inspection Results
C-1	Less than 5% of the total tubes inspected are degraded tubes and none of the inspected tubes are defective.
C-2	One or more tubes, but not more than 1% of the total tubes inspected are defective, or between 5% and 10% of the total tubes inspected are degraded tubes.
C-3	More than 10% of the total tubes inspected are degraded tubes or more than 1% of the inspected tubes are defective.

----------------------NOTE------------------
In all inspections, previously degraded tubes must exhibit significant (greater than 10%) further wall penetrations to be included in the above percentage calculations.

(continued)

5.5 Programs and Manuals (continued)

5.5.9.2b Steam Generator Tube Sleeve Sample Selection and Inspection - The steam generator tube sleeve minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Table 5.5.9-3. The inservice inspection of steam generator tube sleeves shall be performed at the frequencies specified in 5.5.9.3 and the inspected tubes shall be verified acceptable per the acceptance criteria of 5.5.9.4. The tube sleeves selected for each inservice inspection shall include at least 20% of the total number of tube sleeves in all steam generators; the tube sleeves selected for these inspections shall be selected on a random basis except:

a. Where experience in similar plants with similar water chemistry indicates critical areas to be inspected, then at least 50% of the tube sleeves inspected shall be from these critical areas. Where the number of sleeves in the critical areas represents less than 50% of the initial sample, all sleeves in the critical areas shall be inspected.

b. The first sample of tube sleeves selected for each inservice inspection (subsequent to the preservice inspection) of each steam generator shall include:

1. All tube sleeves that previously had detectable wall penetrations (greater than 20%).

2. Tube sleeves in those areas where experience has indicated potential problems.

3. A tube sleeve inspection (pursuant to Specification 5.5.9.4.a.8.) shall be performed on each selected tube sleeve. If any selected tube sleeve does not permit the passage of the eddy current probe for a tube sleeve inspection, this shall be recorded and an adjacent tube sleeve shall be selected and subjected to a tube sleeve inspection.

The results of each inspection sample shall be classified into one of the following three categories (this classification shall apply to the inspection of sleeves and treated exclusive of the tube inspections in 5.5.9.2a):

(continued)

5.5 Programs and Manuals (continued)

5.5.9.2b Steam Generator Tube Sleeve Sample Selection and Inspection (continued)

Category	Inspection Results
C-1	Less than 5% of the total tube sleeves inspected are degraded tube sleeves and none of the inspected tube sleeves are defective.
C-2	One or more tube sleeves, but not more than 1% of the total tube sleeves inspected are defective, or between 5% and 10% of the total tube sleeves inspected are degraded tube sleeves.
C-3	More than 10% of the total tube sleeves inspected are degraded tube sleeves or more than 1% of the inspected tube sleeves are defective.

----------------------NOTE------------------

In all inspections, previously degraded tube sleeves must exhibit significant (greater than 10%) further wall penetrations to be included in the above percentage calculations.

5.5.9.3 Inspection Frequencies - The above required inservice inspections of steam generator tubes and tube sleeves shall be performed at the following frequencies:

a. The first inservice inspection of the steam generator tubes shall be performed after 6 Effective Full Power Months but within 24 calendar months of initial criticality. Subsequent inservice inspections shall be performed at intervals of not less than 12 nor more than 24 calendar months after the previous inspection. If two consecutive inspections following service under AVT conditions, not including the preservice inspection, result in all inspection results falling into the C-1 category or if two consecutive inspections demonstrate that previously observed degradation has not continued and no additional degradation has occurred, the inspection interval may be extended to a maximum of once per 40 months.

(continued)

5.5 Programs and Manuals (continued)

5.5.9.3 Inspection Frequencies (continued)

b. The inservice inspection of steam generator tube sleeves shall be performed at the following frequencies:

1. Steam generator tube sleeves shall be inspected prior to initial operation. The operating period before the initial inservice inspection shall not be shorter than six months nor longer than 24 months. The inspections of tube sleeves shall be configured to ensure that each individual tube sleeve is inspected at least once in 60 months.

2. If the results of the inservice inspection of steam generator tube sleeves conducted in accordance with Table 5.5.9-3 fall in category C-3, the inspection frequency shall be increased to ensure that each remaining tube sleeve is inspected at least once in 30 months. The increase in inspection frequency shall apply until the subsequent inspection satisfies the criteria for Category C-1.

c. If the results of the inservice inspection of a steam generator conducted in accordance with Tables 5.5.9-2 and 5.5.9-3 at 40 month intervals fall into Category C-3, the inspection frequency for the applicable tube or sleeve inspection shall be increased to at least once per 20 months. The increase in inspection frequency shall apply until the subsequent inspections satisfy the criteria of Specification 5.5.9.3.a (tubes) or 5.5.9.3.b.3 (sleeves); the interval may then be extended to a maximum of once per 40 months (tubes) or 30 months (sleeves).

d. Additional, unscheduled inservice inspections shall be performed on each steam generator in accordance with the first sample inspection specified in Tables 5.5.9-2 and 5.5.9-3 during the shutdown subsequent to any of the following conditions:

1. Primary-to-secondary tubes leaks (not including leaks originating from tube-to-tube sheet welds) in excess of the limits of Specification 3.4.14.

2. A seismic occurrence greater than the Operating Basis Earthquake.

(continued)

5.5 Programs and Manuals (continued)

5.5.9.3 Inspection Frequencies (continued)

3. A loss-of-coolant accident requiring actuation of the engineered safeguards.

4. A main steam line or feedwater line break.

5.5.9.4 Acceptance Criteria

a. As used in this Specification

1. Defect means an imperfection of such severity that it exceeds the plugging or repair limit. A tube or sleeve containing a defect is defective.

2. Degradation means a service-induced cracking, wastage, wear, or general corrosion occurring on either inside or outside of a tube.

3. % Degradation means the percentage of the tube wall thickness affected or removed by degradation.

4. Degraded Tube or Sleeve means a tube or sleeve containing imperfections greater than or equal to 20% of the nominal wall thickness caused by degradation.

5. Imperfection means an exception to the dimensions, finish, or contour of a tube from that required by fabrication drawings or specifications. Eddy-current testing indications below 20% of the nominal tube wall thickness, if detectable, may be considered as imperfections.

6. Plugging or Repair Limit means the imperfection depth at or beyond which the tube shall be removed from service by plugging or repaired by sleeving in the affected area. The plugging or repair imperfection depths specified below are in percentage of nominal wall thickness:

a.	Original tube wall	40%
b.	ABB-CE leak tight sleeve wall	35%

(continued)

5.5 Programs and Manuals (continued)

5.5.9.4 Acceptance Criteria (continued)

7. Preservice Inspection in the context of new steam generators means an inspection of the full length of each tube in each steam generator performed by eddy current techniques prior to service to establish a baseline condition of the tubing. This inspection was performed prior to the field hydrostatic test and prior to initial POWER OPERATION using the equipment and techniques expected to be used during subsequent inservice inspections.

   Preservice Inspection for steam generator tubes repaired by tube sleeving means an inspection of the full length of the pressure boundary portion of the sleeved area performed by eddy current techniques prior to service to establish a baseline condition of the sleeved area. The sleeved area includes the pressure retaining portions of the parent tube in contact with the sleeve, the sleeve-to-tube weld and the pressure retaining portion of the sleeve.

8. Sleeve Inspection for sleeves selected in accordance with table 5.5.9-3 means an inspection of the sleeved area, including the pressure retaining portions of the parent tube in contact with the sleeve, the sleeve-to-tube weld and the pressure retaining portion of the sleeve.

9. Tube or Tubing means that portion of the tube that forms the primary system to secondary system pressure boundary.

10. Tube Inspection for tubes selected in accordance with Table 5.5.9-2 means an inspection of the steam generator tube from the point of entry (hot leg side) completely around the U-bend to the top support of the cold leg, excluding sleeved areas.

(continued)

5.5 Programs and Manuals (continued)

5.5.9.4 Acceptance Criteria (continued)

11. Tube Repair or Sleeve refers to welded sleeving, as described in Combustion Engineering, Inc. (CE or ABB-CE) report CEN-630-P, "Repair of 3/4" O.D. Steam Generator Tubes Using Leak Tight Sleeves," Revision 02, June 1997, which is used to maintain a tube in service or to return a previously plugged tube to service. Returning a previously plugged tube to service includes the removal of the tube plugs that were installed as a preventive or corrective measure and performing a tube inspection of the tube in accordance with Technical Specification 5.5.9.4.a.8.

12. Unserviceable describes the condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of an Operating Basis Earthquake, a loss-of-coolant accident, or a steam line or feedwater line break as specified in 5.5.9.3.d., above.

b. The steam generator shall be determined OPERABLE after completing the corresponding actions required by Tables 5.5.9-2 and 5.5.9-3, including the following:

1. Plug or repair all defective tubes and all tubes containing through-wall cracks.

2. Plug all tubes containing any defective sleeves and all tubes containing any sleeves with through-wall cracks.

(continued)

TABLE 5.5.9-1
MINIMUM NUMBER OF STEAM GENERATORS TO BE INSPECTED DURING INSERVICE INSPECTION

Preservice Inspection	No	Yes
No. of Steam Generators per Unit	Two	Two
First Inservice Inspection	All	One
Second & Subsequent Inservice Inspection	One*	One*

TABLE NOTATION

*The inservice inspection may be limited to one steam generator on a rotating schedule encompassing 3 N % of the tubes (where N is the number of steam generators in the plant) if the results of the first or previous inspections indicate that all steam generators are performing in a like manner. Note that under some circumstances, the operating conditions in one or more steam generators may be found to be more severe than those in other steam generators. Under such circumstances the sample sequence shall be modified to inspect the most severe conditions.

TABLE 5.5.9-2
STEAM GENERATOR TUBE INSPECTION

1ST SAMPLE INSPECTION			2ND SAMPLE INSPECTION		3RD SAMPLE INSPECTION	
Sample Size	Result	Action Required	Result	Action Required	Result	Action Required
A minimum of S Tubes per S.G	C-1	None	N.A.	N.A.	N.A.	N.A.
	C-2	Plug or repair defective tubes and inspect additional 2S tubes in this S.G.	C-1	None	N.A.	N.A.
			C-2	Plug or repair defective tubes and inspect additional 4S tubes in this S.G.	C-1	None
					C-2	Plug or repair defective tubes
					C-3	Perform action for C-3 result of first sample
			C-3	Perform action for C-3 result of first sample	N.A.	N.A.
	C-3	Inspect all tubes in this S.G., plug or repair defective tubes and inspect 2S tubes in each other S.G. Notification to NRC pursuant to 10 CFR 50.72 (b)(2)	All other S.G.s are C-1	None	N.A.	N.A.
			Some S.G.s C-2 but no additional S.G. are C-3	Perform action for C-2 result of second sample	N.A.	N.A.
			Additional S.G. is C-3	Inspect all tubes in each S.G. and plug or repair defective tubes. Notification to NRC pursuant to 10 CFR 50.72 (b)(2)	N.A.	N.A.

$S = 3\frac{N}{n}\%$ Where N is the number of steam generators in the unit, and n is the number of steam generators inspected during an inspection.

TABLE 5.5.9-3
STEAM GENERATOR SLEEVE INSPECTION

1ST SAMPLE INSPECTION			2ND SAMPLE INSPECTION	
Sample Size	Result	Action Required	Result	Action Required
A minimum of 20% of the sleeves per S.G.	C-1	None	N.A.	N.A.
	C-2	Plug tubes containing defective sleeves and inspect all remaining installed sleeves in this S.G.	C-1	None
			C-2	Plug tubes containing defective sleeves
			C-3	Perform action for C-3 result of first sample
	C-3	Inspect all installed sleeves in this S.G., plug tubes containing defective sleeves and inspect 100% of the installed sleeves in the other S.G. Notification to NRC pursuant to 10 CFR 50.72 (b)(2)	Other S.G. is C-1	None
			Other S.G. is C-2	Plug tubes containing defective sleeves
			Other S.G. is C-3	Inspect all sleeves in each S.G. and plug tubes containing defective sleeves. Notification to NRC pursuant to 10 CFR 50.72(b)(2)

5.5 Programs and Manuals (continued)

5.5.10 Secondary Water Chemistry Program

This program provides controls for monitoring secondary water chemistry to inhibit SG tube degradation and low pressure turbine disc stress corrosion cracking. The program shall include:

a. Identification of a sampling schedule for the critical variables and control points for these variables;

b. Identification of the procedures used to measure the values of the critical variables;

c. Identification of process sampling points which shall include monitoring the discharge of the condensate pumps for evidence of condenser in leakage;

d. Procedures for the recording and management of data;

e. Procedures defining corrective actions for all off control point chemistry conditions; and

f. A procedure identifying the authority responsible for the interpretation of the data and the sequence and timing of administrative events, which is required to initiate corrective action.

5.5.11 Ventilation Filter Testing Program (VFTP)

A program shall be established to implement the following required testing of Engineered Safety Feature (ESF) filter ventilation systems at the frequencies specified in Regulatory Guide 1.52, Revision 2, and in accordance with Regulatory Guide 1.52, Revision 2 and ANSI N510-1980 at the system flowrate specified below ± 10%.

a. Demonstrate for each of the ESF systems that an inplace test of the high efficiency particulate air (HEPA) filters shows a penetration and system bypass ≤ 1.0 % when tested in accordance with Regulatory Guide 1.52, Revision 2, and ANSI N510-1980, at the system flowrate specified as follows ± 10%:

(continued)

5.5 Programs and Manuals (continued)

5.5.11 Ventilation Filter Testing Program (VFTP) (continued)

ESF Ventilation System	Flowrate
Control Room Essential Filtration System (CREFS)	28,600 CFM
Engineered Safety Feature (ESF) Pump Room Exhaust Air Cleanup System (PREACS)	6,000 CFM

b. Demonstrate for each of the ESF systems that an inplace test of the charcoal adsorber shows a penetration and system bypass ≤ 1.0 % when tested in accordance with Regulatory Guide 1.52, Revision 2, and ANSI N510-1980 at the system flowrate specified as follows ± 10%:

ESF Ventilation System	Flowrate
CREFS	28,600 CFM
ESF PREACS	6,000 CFM

c. Demonstrate for each of the ESF systems that a laboratory test of a sample of the charcoal adsorber, when obtained as described in Regulatory Guide 1.52, Revision 2, and ANSI N510-1980 shows the methyl iodide penetration less than the value specified below when tested in accordance with ASTM D3803-1979 at a temperature of 80°C ± 0.5°C and greater than or equal to the relative humidity specified as follows:

ESF Ventilation System	Penetration	RH
CREFS	≤ 1.0%	70%
ESF PREACS	≤ 1.0%	70%

(continued)

5.5 Programs and Manuals (continued)

5.5.11 Ventilation Filter Testing Program (VFTP) (continued)

d. For each of the ESF systems, demonstrate the pressure drop across the combined HEPA filters, the prefilters, and the charcoal adsorbers is less than the value specified below when tested in accordance with Regulatory Guide 1.52, Revision 2, and ANSI N510-1980 at the system flowrate specified as follows ± 10%:

ESF Ventilation System	Delta P	Flowrate
CREFS	8.4 inches water gauge	28,600 CFM
ESF PREACS	8.4 inches water gauge	6,000 CFM

e. Demonstrate that the heaters for each of the ESF systems dissipate the following specified value when tested in accordance with ANSI N510-1980:

ESF Ventilation System	Wattage
ESF PREACS	> 19 kW

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the VFTP test frequencies.

5.5.12 Explosive Gas and Storage Tank Radioactivity Monitoring Program

This program provides control for potentially explosive gas mixtures contained in the Waste Gas Holdup System, the quantity of radioactivity contained in gas storage tanks and the quantity of radioactivity contained in unprotected outdoor liquid storage tanks. The gaseous radioactivity quantities shall be determined following the methodology in Branch Technical Position (BTP) ETSB 11-5, "Postulated Radioactive Release due to Waste Gas System Leak or Failure". The liquid radwaste quantities shall be determined in accordance with Standard Review Plan, Section 15.7.3, "Postulated Radioactive Release due to Tank Failures".

(continued)

5.5 Programs and Manuals (continued)

5.5.12 Explosive Gas and Storage Tank Radioactivity Monitoring Program (continued)

The program shall include:

a. The limits for concentrations of hydrogen and oxygen in the Waste Gas Holdup System and a surveillance program to ensure the limits are maintained. Such limits shall be appropriate to the system's design criteria (i.e., whether or not the system is designed to withstand a hydrogen explosion);

b. A surveillance program to ensure that the quantity of radioactivity contained in each gas storage tank is less than the amount that would result in a whole body exposure of ≥ 0.5 rem to any individual in an unrestricted area, in the event of an uncontrolled release of the tanks' contents; and

c. A surveillance program to ensure that the quantity of radioactivity contained in all outdoor liquid radwaste tanks that are not surrounded by liners, dikes, or walls, capable of holding the tanks' contents and that do not have tank overflows and surrounding area drains connected to the Liquid Radwaste Treatment System is less than the amount that would result in concentrations less than the limits of 10 CFR Part 20, Appendix B, Table 2, Column 2, at the nearest potable water supply and the nearest surface water supply in an unrestricted area, in the event of an uncontrolled release of the tanks' contents.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Explosive Gas and Storage Tank Radioactivity Monitoring Program surveillance frequencies.

5.5.13 Diesel Fuel Oil Testing Program

A diesel fuel oil testing program to implement required testing of both new fuel oil and stored fuel oil shall be established. The program shall include sampling and testing requirements, and acceptance criteria, all in accordance with applicable ASTM Standards as referenced in the UFSAR. The purpose of the program is to establish the following:

(continued)

5.5 Programs and Manuals (continued)

5.5.13 Diesel Fuel Oil Testing Program (continued)

a. Acceptability of new fuel oil for use prior to addition to storage tanks by determining that the fuel oil has:

   1. An API gravity or an absolute specific gravity within limits,

   2. A flash point and kinematic viscosity within limits for ASTM 2D fuel oil, and

   3. Water and sediment are within the limits of ASTM D1796;

b. Other properties for ASTM 2D fuel oil are within limits within 31 days following sampling and addition to storage tanks; and

c. Total particulate concentration of the stored fuel oil is ≤ 10 mg/l when tested every 92 days in accordance with ASTM D-2276, Method A-2 or A-3.

5.5.14 Technical Specifications (TS) Bases Control Program

This program provides a means for processing changes to the Bases of these Technical Specifications.

a. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.

b. Licensees may make changes to Bases without prior NRC approval provided the changes do not involve either of the following:

   A change in the TS incorporated in the license; or

   A change to the updated FSAR or Bases that involves an unreviewed safety question as defined in 10 CFR 50.59.

c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the FSAR.

(continued)

5.5 Programs and Manuals (continued)

5.5.14 Technical Specifications (TS) Bases Control Program (continued)

d. Proposed changes that meet the criteria of Specification 5.5.14b above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

5.5.15 Safety Functions Determination Program (SFDP)

This program ensures loss of safety function is detected and appropriate actions taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other appropriate limitations and remedial or compensatory actions may be identified to be taken as a result of the support system inoperability and corresponding exception to entering supported system Condition and Required Actions. This program implements the requirements of LCO 3.0.6. The SFDP shall contain the following:

a. Provisions for cross train checks to ensure a loss of the capability to perform the safety function assumed in the accident analysis does not go undetected;

b. Provisions for ensuring the plant is maintained in a safe condition if a loss of function condition exists;

c. Provisions to ensure that an inoperable supported system's Completion Time is not inappropriately extended as a result of multiple support system inoperabilities; and

d. Other appropriate limitations and remedial or compensatory actions.

A loss of safety function exists when, assuming no concurrent single failure, a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and:

a. A required system redundant to system(s) supported by the inoperable support system is also inoperable; or

(continued)

5.5 Programs and Manuals (continued)

5.5.15 Safety Functions Determination Program (continued)

b. A required system redundant to system(s) in turn supported by the inoperable supported system is also inoperable; or

c. A required system redundant to support system(s) for the supported systems (a) and (b) above is also inoperable.

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

5.5.16 Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September, 1995, as modified by the following exceptions:

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 52.0 psig. The containment design pressure is 60 psig.

The maximum allowable containment leakage rate, L_a, at P_a, shall be 0.1 % of containment air weight per day.

Leakage Rate acceptance criteria are:

a. Containment leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance are $< 0.60\ L_a$ for the Type B and C tests and $\leq 0.75\ L_a$ for Type A tests.

b. Air lock testing acceptance criteria are:

1. Overall air lock leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$.

2. For each door, leakage rate is $\leq 0.01\ L_a$ when pressurized to $\geq$ 14.5 psig.

(continued)

5.5 Programs and Manuals (continued)

5.5.16 Containment Leakage Rate Testing Program (continued)

The provisions of SR 3.0.2 do not apply to the test frequencies in the Containment Leakage Rate Testing Program.

The provisions of SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program.

5.0 ADMINISTRATIVE CONTROLS

5.6 Reporting Requirements

The following reports shall be submitted in accordance with 10 CFR 50.4.

5.6.1 Occupational Radiation Exposure Report

--------------------------------NOTE--------------------------------

A single submittal may be made for a multiple unit station. The submittal should combine sections common to all units at the station.

A tabulation on an annual basis of the number of station, utility, and other personnel (including contractors) receiving exposures > 100 mrem/yr and their associated man rem exposure according to work and job functions (e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance, waste processing, and refueling). This tabulation supplements the requirements of 10 CFR 20.2206. The dose assignments to various duty functions may be estimated based on pocket dosimeter, thermoluminescent dosimeter (TLD), or film badge measurements. Small exposures totalling < 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole body dose received from external sources should be assigned to specific major work functions. The report shall be submitted by April 30 of each year.

5.6.2 Annual Radiological Environmental Operating Report

--------------------------------NOTE--------------------------------

A single submittal may be made for a multiple unit station. The submittal should combine sections common to all units at the station.

The Annual Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted by May 15 of each year. The report shall include summaries, interpretations, and analyses of trends of the results of the radiological environmental monitoring program for the reporting period. The material provided shall be consistent with the objectives outlined in the Offsite Dose Calculation Manual (ODCM), and in 10 CFR 50, Appendix I, Sections IV.B.2, IV.B.3, and IV.C.

(continued)

5.6 Reporting Requirements (continued)

5.6.2 Annual Radiological Environmental Operating Report (continued)

The Annual Radiological Environmental Operating Report shall include the results of analyses of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the table and figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements in the format of the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979. The report shall identify the TLD results that represent collocated dosimeters in relation to the NRC TLD program and the exposure period associated with each result. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted in a supplementary report as soon as possible.

5.6.3 Radioactive Effluent Release Report

-------------------------------NOTE-------------------------------
A single submittal may be made for a multiple unit station. The submittal should combine sections common to all units at the station; however, for units with separate radwaste system, the submittal shall specify the releases of radioactive material from each unit.

The Radioactive Effluent Release Report covering the operation of the unit shall be submitted in accordance with 10 CFR 50.36a. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be consistent with the objectives outlined in the ODCM and Process Control Program and in conformance with 10 CFR 50.36a and 10 CFR 50, Appendix I, Section IV.B.1.

5.6.4 Monthly Operating Reports

Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the shutdown cooling system suction line relief valves or pressurizer safety valves, shall be submitted on a monthly basis no later than the 15th of each month following the calendar month covered by the report.

(continued)

5.6 Reporting Requirements (continued)

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

1. Shutdown Margin - Reactor Trip Breakers Open for Specification 3.1.1.

2. Shutdown Margin - Reactor Trip Breakers Closed for Specification 3.1.2.

3. Moderator Temperature Coefficient BOL and EOL limits for Specification 3.1.4.

4. Boron Dilution Alarm System for Specification 3.3.12.

5. CEA Alignment for Specification 3.1.5.

6. Regulating CEA Insertion Limits for Specification 3.1.7.

7. Part Length CEA Insertion Limits for Specification 3.1.8.

8. Linear Heat Rate for Specification 3.2.1.

9. Azimuthal Power Tilt - T_q for Specification 3.2.3.

10. DNBR for Specification 3.2.4.

11. Axial Shape Index for Specification 3.2.5.

12. Boron Concentration (Mode 6) for Specification 3.9.1.

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

1. "CE Method for Control Element Assembly Ejection Analysis, "CENPD-0190-A, January 1976 (Methodology for Specification 3.1.7, Regulating CEA Insertion Limits).

(continued)

5.6 Reporting Requirements (continued)

5.6.5 Core Operating Limits Report (COLR) (continued)

2. "The ROCS and DIT Computer Codes for Nuclear Design," CENPD-266-P-A, April 1983 [Methodology for Specifications 3.1.1, Shutdown Margin - Reactor Trip Breakers Open; 3.1.2, Shutdown Margin - Reactor Trip Breakers Closed; 3.1.4, Moderator Temperature Coefficient BOL and EOL limits; 3.1.7, Regulating CEA Insertion Limits and 3.9.1, Boron Concentration (Mode 6)].

3. "Safety Evaluation Report related to the Final Design of the Standard Nuclear Steam Supply Reference Systems CESSAR System 80, Docket No. STN 50-470, "NUREG-0852 (November 1981), Supplements No. 1 (March 1983), No. 2 (September 1983), No. 3 (December 1987) [Methodology for Specifications 3.1.2, Shutdown Margin - Reactor Trip Breakers Closed; 3.1.4, Moderator Temperature Coefficient BOL and EOL limits; 3.3.12, Boron Dilution Alarm System; 3.1.5, CEA Alignment; 3.1.7, Regulating CEA Insertion Limits; 3.1.8, Part Length CEA Insertion Limits and 3.2.3, Azimuthal Power Tilt - T_q].

4. "Modified Statistical Combination of Uncertainties," CEN-356(V)-P-A Revision 01-P-A, May 1988 and "System 80™ Inlet Flow Distribution," Supplement 1-P to Enclosure 1-P to LD-82-054, February 1993 (Methodology for Specification 3.2.4, DNBR and 3.2.5 Axial Shape Index).

5. "Calculative Methods for the CE Large Break LOCA Evaluation Model for the Analysis of CE and W Designed NSSS," CENPD-132, Supplement 3-P-A, June 1985 (Methodology for Specification 3.2.1, Linear Heat Rate).

6. "Calculative Methods for the CE Small Break LOCA Evaluation Model," CENPD-137-P, August 1974 (Methodology for Specification 3.2.1, Linear Heat Rate).

7. "Calculative Methods for the CE Small Break LOCA Evaluation Model," CENPD-137-P, Supplement 1P, January 1977 (Methodology for Specification 3.2.1, Linear Heat Rate).

(continued)

5.6 Reporting Requirements (continued)

5.6.5 Core Operating Limits Report (COLR) (continued)

8. Letter: O.D. Parr (NRC) to F. M. Stern (CE), dated June 13, 1975 (NRC Staff Review of the Combustion Engineering ECCS Evaluation Model). NRC approval for: 5.6.5.b.6.

9. Letter: K. Kniel (NRC) to A. E. Scherer (CE), dated September 27, 1977 (Evaluation of Topical Reports CENPD-133, Supplement 3-P and CENPD-137, Supplement 1-P). NRC approval for 5.6.5.b.7.

10. "Fuel Rod Maximum Allowable Pressure," CEN-372-P-A, May 1990 (Methodology for Specification 3.2.1, Linear Heat Rate).

11. Letter: A. C. Thadani (NRC) to A. E. Scherer (CE), dated April 10, 1990, ("Acceptance for Reference CE Topical Report CEN-372-P"). NRC approval for 5.6.5.b.10.

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any mid cycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

5.6.6 PAM Report

When a report is required by Condition B or G of LCO 3.3.10, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

(continued)

5.6 Reporting Requirements (continued)

5.6.7 Tendon Surveillance Report

Any abnormal degradation of the containment structure detected during the tests required by the Pre-Stressed Concrete Containment Tendon Surveillance Program shall be reported to the NRC within 30 days. The report shall include a description of the tendon condition, the condition of the concrete (especially at tendon anchorages), the inspection procedures, the tolerances on cracking, and the corrective action taken.

5.6.8 Steam Generator Tube Inspection Report

Within 15 days following the completion of each inservice inspection of steam generator tubes, the number of tubes plugged and/or repaired in each steam generator shall be reported to the Commission in a Special Report.

The complete results of the steam generator tube inservice inspection shall be submitted to the Commission in a Special Report within 12 months following completion of the inspection. This Special Report shall include:

a. Number and extent of tubes inspected.

b. Location and percent of wall-thickness penetration for each indication of an imperfection.

c. Identification of tubes plugged and/or repaired.

Results of steam generator tube and sleeve inspections which fall into Category C-3 shall be reported in a Special Report to the Commission within 30 days and prior to resumption of plant operation and shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

5.0 ADMINISTRATIVE CONTROLS

5.7 High Radiation Area

5.7.1 In addition to the provisions of 10 CFR 20.1601, the following controls provide an alternate method for controlling access to high radiation areas as provided by paragraph 20.1601(c) of 10 CFR part 20. High radiation areas, as defined in 10 CFR 20, in which the intensity of radiation is > 100 mrem/hr but ≤ 1000 mrem/hr, shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Exposure Permit (REP). Individuals qualified in radiation protection procedures (e.g., Radiation Protection Technicians) or personnel continuously escorted by such individuals may be exempt from the REP issuance requirement during the performance of their assigned duties in high radiation areas with exposure rates ≤ 1000 mrem/hr, provided they are otherwise following plant radiation protection procedures for entry into such high radiation areas.

Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

a. A radiation monitoring device that continuously indicates the radiation dose rate in the area.

b. A radiation monitoring device that continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate levels in the area have been established and personnel are aware of them.

c. An individual qualified in radiation protection procedures with a radiation dose rate monitoring device, who is responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by the Radiation Protection Section Leader or designated alternate in the REP.

(continued)

5.7 High Radiation Area

5.7.2 In addition to the requirements of Specification 5.7.1, areas accessible to personnel with radiation levels such that an individual could receive in 1 hour a dose greater than 1000 mrem shall be provided with locked or continuously guarded doors to prevent unauthorized entry and the keys shall be maintained under the administrative control of the Shift Manager on duty or Radiation Protection supervision. Doors shall remain locked except during periods of access by personnel under an approved REP that shall specify the dose rate levels in the immediate work areas and the maximum allowable stay times for individuals in those areas. In lieu of the stay time specification of the REP, direct or remote (such as closed circuit TV cameras) continuous surveillance may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities being performed within the area.

5.7.3 For individual high radiation areas accessible to personnel with radiation levels such that an individual could receive in 1 hour a dose in excess of 1000 mrem (measurement made at 30 cm from source of radioactivity), that are located within large areas such as reactor containment, where no enclosure exists for purposes of locking, and where no enclosure can be reasonably constructed around the individual area, that individual area shall be barricaded and conspicuously posted, and a flashing light shall be activated as a warning device.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 120 TO FACILITY OPERATING LICENSE NO. NPF-41,

AMENDMENT NO. 120 TO FACILITY OPERATING LICENSE NO. NPF-51,

AND AMENDMENT NO. 120 TO FACILITY OPERATING LICENSE NO. NPF-74

ARIZONA PUBLIC SERVICE COMPANY, ET AL.

PALO VERDE NUCLEAR GENERATING STATION, UNITS 1, 2, AND 3

DOCKET NOS. STN 50-528, STN 50-529, AND STN 50-530

1.0 INTRODUCTION

By application dated May 23, 1997, as supplemented by letters dated September 27, 1998, and May 26, 1999, the Arizona Public Service Company (APS or the licensee) requested changes to the Technical Specifications (TS) for the Palo Verde Nuclear Generating Station (Palo Verde), Units 1, 2, and 3. APS submitted this request on behalf of itself, the Salt River Project Agricultural Improvement and Power District, Southern California Edison Company, El Paso Electric Company, Public Service Company of New Mexico, Los Angeles Department of Water and Power, and Southern California Public Power Authority. The proposed changes would allow the installation of ABB Combustion Engineering (ABB-CE) leak-tight sleeves in defective steam generator tubes as a tube repair method.

2.0 BACKGROUND

The ABB-CE sleeves consist of a tubesheet sleeve design and a tube support sleeve design. A tubesheet sleeve is designed to repair the degraded portion of a tube in the vicinity of the top of the tubesheet. A tube support sleeve is designed to repair the degraded freespan or support plate region of a tube. The licensee-proposed sleeve repair method is based on the report, "Repair of 3/4-inch O. D. [Outer Diameter] Steam Generator Tubes Using Leak Tight Sleeves," CEN-630-P, Revision 02, June 1997 (Proprietary information. Not publicly available. A nonproprietary version of CEN-630 was submitted to NRC on September 16, 1997).

The staff has approved the use of similarly designed sleeves in U.S. nuclear plants. The staff review of the licensee's submittal is therefore focused on those issues warranting revision, amplification, or inclusion based on recent field experience. Details of prior staff evaluations of ABB-CE leak-tight sleeves may be found in the safety evaluations for Waterford Steam Electric Station, Unit 3, docket number 50-382, dated December 14, 1995; Byron Nuclear Power Station, Units 1 and 2 and Braidwood Nuclear Power Station, Units 1 and 2, docket numbers 50-454, 50-455, 50-456, and 50-457, dated April 12, 1996; Kewaunee Nuclear Power Plant, docket No.

50-305, dated June 7, 1997; Prairie Island Units 1 and 2, docket numbers 50-282 and 50-306, dated November 4, 1997; Beaver Valley Unit 1, docket number 50-334, dated November 25, 1997; and San Onofre Units 2 and 3, docket numbers 50-361 and 50-362, dated August 26, 1998.

Previous staff evaluation of ABB-CE sleeves addressed the technical adequacy of the sleeves in the four principal areas of pressure-retaining component design: structural requirements, material of construction, welding, and nondestructive examination. The staff found the analyses and tests that were submitted to address these areas of component design to be acceptable.

The function of sleeves is to restore the structural and leakage integrity of the tube pressure boundary. Consequently, structural analyses were performed for a variety of loadings including design pressure, operating transients, and other parameters selected to envelop loads imposed during normal operating, upset, and accident conditions. Stress analyses of sleeved tube assemblies were performed in accordance with the requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section III. As described in detail in CEN-630-P, Revision 02, the structural integrity of the sleeve design has been investigated analytically and verified by laboratory tests of sleeve mockups. These analyses, along with the results of qualification testing and previous plant operating experience, were cited to demonstrate that the sleeved tube assembly is capable of restoring steam generator tube integrity.

The sleeve material, thermally treated (TT) Alloy 690, is a nickel-iron-chromium alloy. It is an ASME Code-approved material, specified in ASME SB-163, and is incorporated in ASME Code Case N-20. The staff has determined that the use of Alloy 690 TT material is an improvement over the Alloy 600 material used in the parent tube. Corrosion tests conducted under Electric Power Research Institute (EPRI) sponsorship confirm that Alloy 690 TT material resists corrosion better than that of Alloy 600. As a result of these laboratory corrosion tests, the staff has concluded that Alloy 690 TT material satisfies the guidelines in Regulatory Guide (RG) 1.85, "Materials Code Case Acceptability ASME Section III, Division 1," Revision 24, dated July 1986. The staff has approved use of Alloy 690 TT tubing in previous sleeving applications.

For the tubesheet sleeve, the upper end of the sleeve is welded to the parent tube in the freespan region above the tubesheet and the lower end is hard-rolled into the tubesheet below the expansion zone. For the tube support sleeve, both ends of the sleeve are welded to the parent tube. The welding process uses automatic autogenous gas tungsten arc welding which was qualified and demonstrated during laboratory tests by full-scale mock-ups. Qualification of the welding procedures and welding equipment operator was performed in accordance with the specifications of the ASME Code, Section IX.

The staff considers sleeves to be a long-term repair but not a repair with unlimited service life. The welding of the sleeve to the tube may create new locations susceptible to stress corrosion cracking and the time for the initiation of service-induced degradation in sleeve-tube assemblies is difficult to quantify. The staff finds the accelerated corrosion tests of sleeve-tube assemblies, conducted by the vendors to predict service life, not reliable for deterministic predictions. In order to get a more realistic evaluation of the degradation, the licensees inspect a sample of sleeves at each outage to ensure that any sleeve degradation is detected and addressed early.

This results in a better and more accurate prediction of the sleeve life. The inservice inspection requirements for the sleeve inspection are discussed further in Section 3.2 of this safety evaluation. The staff considers the corrosion tests coupled with the inservice inspection of the sleeves at each outage to give a viable indicator of potential performance and therefore acceptable.

3.0 EVALUATION

Experience with all types of steam generator tube sleeves has revealed certain issues that need to be evaluated in addition to sleeve design and qualification as discussed in previous NRC safety evaluations. These issues involve weld preparation, weld acceptance inspections, inservice inspection expansion criteria, sleeve plugging limits, post-weld heat treatment, and primary-to-secondary leakage limits, which are discussed below.

3.1 Weld Preparation and Acceptance Inspections

During its spring 1996 refueling outage, the licensee of a domestic nuclear plant detected eddy current test (ET) indications in about 60 weld joints in ABB-CE leak-tight sleeves. This finding was the result of using a new, more sensitive ET probe. The ET indications were caused by entrapped oxides and/or weld shrinkage within the sleeve-to-tube weld. The cause of these weld defects was traced to an inadequate tube cleaning process. Although the defects did not significantly impair the structural integrity (strength) of the welds and did not cause leakage, they did increase the probability of leakage. In a separate case, during an installation of welded sleeves in another domestic nuclear plant, weld zone indications were identified visually but were not detected by either ET or ultrasonic testing (UT). These findings pointed to the inadequacy of previous sleeve installation and inspection.

ABB-CE has revised its weld preparation procedures and incorporated these changes in CEN-630-P, Revision 02. Prior to installing a sleeve, the inner surface of the parent tube at the desired weld location is cleaned of service-induced oxides using motorized wire brushes. ABB-CE specifies that after surface cleaning, every repaired tube be visually inspected to confirm adequate surface cleaning. ABB-CE advises that the visual inspection of every tube is an interim measure until sufficient field experience is gained to consider adoption of statistical sampling in the future.

ABB-CE has also revised its procedures for weld acceptance inspection. The initial weld acceptance inspection, performed by UT, was revised to give greater sensitivity. An optional visual inspection, the VT-1 inspection process specified in ASME Code Section XI, was added to the inspection procedure. The initial baseline ET, normally used only as a reference for future inservice inspections, was modified to supplement the UT as a part of the weld acceptance inspection. All of these refinements to the sleeve inspection were confirmed using a large number of laboratory samples and field mockups. The refinements have been incorporated into CEN-630-P, Revision 02, and are discussed in detail below.

The original UT procedure for the sleeve weld joint was based upon the absence of a mid-wall reflection. In an acceptable sleeve-to-tube weld, the mid-wall reflection (mid-wall of the fused sleeve and tube) would not appear because no interface would exist. Previous field experience showed that lack of fusion was not detected by the original UT procedure. The lack of fusion was caused by axially oriented oxide inclusions from an inadequately prepared tube surface. In the improved UT procedure, the back wall signal from the outside of the parent tube is also monitored for presence in the fused area. Additionally, the back wall signal strength is examined for excessive attenuation. Attenuation beyond the normal signal strength, along with other signal artifacts, can be used to detect unacceptable welds. ABB-CE tested the enhanced UT procedure and demonstrated that the revised UT procedure is reliable. As stated in CEN-630-P, Revision 02, use of the Plus Point probe is now part of the sleeve weld acceptance criteria. ABB-CE has shown that the Plus Point probe reliably detects these process-induced weld defects and blowholes.

The licensee stated that it will perform the required visual inspection after tube cleaning in accordance with CEN-630-P, Revision 02. It will also conduct UT and ET examinations after the completion of the sleeve-to-tube weld for all installed sleeves in accordance with CEN-630-P, Revision 02. In addition, the licensee will perform a VT-1 inspection of each sleeve-to-tube weld until sufficient data has been obtained with UT and ET techniques to show that these techniques are capable of detecting and resolving uncertainties in the weld joint. Accordingly, the staff finds the proposed weld inspection method acceptable.

3.2 Inservice Inspection Requirements

For inservice inspection of sleeved tubes, the licensee has proposed to perform an initial inspection of 20% of sleeves at each refueling outage. This initial sample size is more than the initial tube sample size of 3% required by the current TS. The minimum sample requirements for tube inspections are established to assess the overall condition of steam generator tubing. The licensee's proposed inspection sampling for sleeved tubes is consistent with the current industry guidance for steam generator sleeve examinations as specified in EPRI report, "Steam Generator Examination Guidelines," TR-107569, Revision 5 (Proprietary. A nonproprietary document is available). The licensee's proposed inspection sampling for sleeved tubes is also consistent with sleeve inspection sampling plans previously approved by the staff and detailed in Section 2.0, paragraph 2, of the safety evaluation. Additionally, the proposed TS will require additional tubes be inspected, if warranted by the tube inspection results.

In view of the above, the staff considers the inservice inspection program for the sleeved tubes adequate to detect degradation in them and, therefore, acceptable.

3.3 Sleeve Plugging Limit

The sleeve plugging limit is defined in the proposed TS as the imperfection depth in the sleeve at or beyond which the sleeved tube shall be removed from service. The sleeve plugging limit is calculated from the minimum acceptable sleeve wall thickness to maintain structural integrity. RG 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes," and ASME Code Section III provide guidance on the calculations. In addition to structural consideration, RG 1.121 also suggests that an allowance for nondestructive evaluation (NDE) uncertainty and postulated growth of degradation be accounted for in the sleeve plugging limit when using NDE to evaluate sleeve degradation. The licensee assumes a 10% allowance for ET uncertainty and a 10% allowance for degradation growth per cycle in its calculations. The licensee calculated a minimum acceptable wall thickness of 54.67% due to structural consideration. After deducting a total allowance of 20%, the licensee specified a sleeve plugging limit of 35% of sleeve wall thickness. The licensee has proposed this requirement in TS 5.5.9.4.a.6. The staff finds that the sleeve plugging limit satisfies the recommendations in RG 1.121, and therefore, is acceptable.

3.4 Post-Weld Heat Treatment

Residual stress is a contributor to stress corrosion cracking in steam generator tubing. The welding of the sleeve to the parent tube will introduce residual stresses in both the sleeve and the tube. These stresses may increase the susceptibility of the welded joints to stress corrosion cracking. A post-weld heat treatment (PWHT) can reduce these stresses and thus may reduce the likelihood of cracking within a welded joint. ABB-CE recommends that a PWHT be a part of the sleeve installation process. The licensee will follow the recommendation in CEN-630-P, Revision 02, in regard to PWHT of the welded joints.

The staff considers the actions planned by the licensee in regard to PWHT adequate to reduce the residual stresses which may in turn reduce the likelihood of cracking within a welded joint and, therefore, acceptable.

3.5 Primary-to-Secondary Leakage Limit

Leak resistance of the sleeve has been demonstrated through laboratory tests. Bounding calculations and laboratory tests have verified that, should leakage develop in the welded or rolled joints of sleeved tubes, it would not exceed 1 gallon per minute (gpm) and, thus, the 10 CFR Part 100 requirements for radiological release would not be affected, even under the most severe postulated conditions. In addition, the licensee has proposed to modify the current primary-to-secondary leakage limit of 500 gallons per day through any one steam generator to the more stringent 150 gallons per day through any one steam generator. This modification is stated in TS 3.4.14.d and is consistent with the operational leakage limit accepted by the staff in other alternate repair reviews. The staff has determined that the primary-to secondary leakage limits verified by laboratory tests and the licensee's proposal to modify the current primary-to secondary leakage limits to more restrictive and stringent values are adequate to keep the radiological releases to within the 10 CFR Part 100 requirements and are therefore acceptable.

3.6 Proposed TS Changes

In order to implement sleeving of the degraded tubes in Palo Verde steam generators, the licensee has proposed the following changes to the plant TS. The changes (other than editorial changes) are summarized below:

TS 3.4.14.d The primary-to-secondary leakage is limited to 150 gallons per day through any one steam generator.

TS 5.5.9 The Steam Generator Tube Surveillance Program is revised to include sleeves.

TS 5.5.9.1 The Steam Generator Sample Selection and Inspection section is revised to include the inspection of sleeves.

TS 5.5.9.2 TS 5.5.9.2.a and 5.5.9.2.b are revised to include sleeve inspection criteria in addition to the existing tube inspection criteria. Table 5.5.9-3 specifies sleeve sample inspection and associated expansion criteria and is referenced in this section.

TS 5.5.9.3 The Inspection Frequency section is revised to include the inspection frequency for sleeves. A requirement is also added to perform a preservice inspection of tubes that have been repaired by sleeving.

TS 5.5.9.4 The Acceptance Criteria section is revised to include sleeves in the definitions of Defect, Degradation, Degraded Sleeve, Plugging or Repair Limit, Preservice Inspection, Sleeve Inspection, and Tube Repair or Sleeve. The plugging limit is specified to be 35% of sleeve wall thickness.

TS 5.5.9.4.11 CEN-630-P, Revision 02, is referenced in this section.

TS 5.6.8 The Steam Generator Tube Inspection Report section requires that specific reports be submitted to NRC when a tube is repaired by sleeving.

The changes contained in the TSs are consistent with the preceding evaluation of the sleeving amendment. The staff concludes that the licensee's proposal on setting the limit on primary-to-secondary leakage, revision of the Steam Generator Tube Surveillance Program to include sleeves, inclusion of the sleeves in the Steam Generator Sample Selection and Inspection section, including the sleeve inspection criteria in addition to the existing tube inspection criteria, and revision of the existing inspection frequency for the sleeves are acceptable.

The staff also reviewed the accompanying editorial changes proposed by the licensee and concludes that they have no substantive effect on plant operation (repagination, changes to section numbers) and are therefore acceptable.

Based on the foregoing, the staff concludes that the licensee may incorporate the proposed changes into Palo Verde Units 1, 2, and 3 TSs.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Arizona State official was notified of the proposed issuance of the amendments. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change requirements with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and change surveillance requirements. The NRC staff has determined that the amendments involve no significant increase in the amounts and no significant change in the types of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (64 FR 32285). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). The amendments also change recordkeeping, reporting, or administrative procedures or requirements. Accordingly, with respect to such changes, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(10). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: J. Tsao

Date: August 5, 1999