operator should be given Datagram Identification Number N1023 and the following message addressed to William H. Bateman, Director, Project Directorate IV–2: petitioner's name and telephone number, date petition was mailed, plant name, and publication date and page number of this Federal Register notice. A copy of the petition should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, and to T. E. Oubre, Esquire, Southern California Edison Company, P. O. Box 800, Rosemead, California 91770, attorney for the licensee.

Nontimely filings of petitions for leave to intervene, amended petitions, supplemental petitions and/or requests for hearing will not be entertained absent a determination by the Commission, the presiding officer or the presiding Atomic Safety and Licensing Board that the petition and/or request should be granted based upon a balancing of the factors specified in 10 CFR 2.714(a)(1)(i)–(v) and 2.714(d).

For further details with respect to this action, see the application for amendment dated June 3, 1996, as superseded by application dated June 25, 1996, which is available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, and at the local public document room located at the Main Library, University of California, P. O. Box 19557, Irvine, California 92713.

Dated at Rockville, Maryland, this 26th day of June 1996.

For the Nuclear Regulatory Commission.

Mel B. Fields,

Project Manager, Project Directorate IV–2, Division of Reactor Projects III/IV, Office of Nuclear Reactor Regulation.

[FR Doc. 96–16877 Filed 7–1–96; 8:45 am]

BILLING CODE 7590–01–P

Sunshine Act Meeting

AGENCY HOLDING THE MEETING: Nuclear Regulatory Commission.

DATE: Weeks of July 1, 8, 15, and 22, 1996.

PLACE: Commissioners' Conference Room, 11555 Rockville Pike, Rockville, Maryland.

STATUS: Public and Closed.

MATTERS TO BE CONSIDERED:

Week of July 1

Tuesday, July 2

10:00 a.m.
Briefing on Alternatives for Regulating Fuel Cycle Facilities (Public Meeting)
(Contact: Ted Sherr, 301–415–7218)

Wednesday, July 3

10:00 a.m.
Briefing on BPR Project on Redesigned Material Licensing Process (Public Meeting)
(Contact: Pat Rathbun, 301–415–7178)
11:30 a.m.
Affirmation Session (Public Meeting) (if needed)

Week of July 8—Tentative

Wednesday, July 10

11:30 a.m.
Affirmation Session (Public Meeting) (if needed)

Week of July 15—Tentative

There are no meetings scheduled for the Week of July 15.

Week of July 22—Tentative

There are no meetings scheduled for the Week of July 22.

ADDITIONAL INFORMATION: By a vote of 3–0 on June 26, the Commission determined pursuant to U.S.C. 552b(e) and § 9.107(a) of the Commission's rules that ''Affirmation of Innovative Weaponry, Inc.—Request for a Hearing'' (Public Meeting) be held on June 26, and on less than one weeks' notice to the public.

The schedule for Commission meetings is subject to change on short notice. To verify the status of meetings call (Recording)—(301) 415–1292.

CONTACT PERSON FOR MORE INFORMATION: Bill Hill (301 415–1661.

The NRC Commission Meeting Schedule can be found on the Internet at:

http://www.nrc.gov/SECY/smj/schedule.htm.

This notice is distributed by mail to several hundred subscribers; if you no longer wish to receive it, or would like to be added to it, please contact the Office of the Secretary, Attn: Operations Branch, Washington, D.C. 20555 (301–415–1963).

In addition, distribution of this meeting notice over the internet system is available. If you are interested in receiving this Commission meeting schedule electronically, please send an electronic message to alb@nrc.gov or dkw@nrc.gov.

Dated: June 28, 1996.

William M. Hill, Jr.,

SECY Tracking Officer, Office of the Secretary.

[FR Doc. 96–17011 Filed 6–28–96; 2:29 pm]

BILLING CODE 7590–01–M

[Docket Nos. 50–528, 50–529 and 50–530]

Arizona Public Service Company; Palo Verde Nuclear Generating Station, Unit Nos. 1, 2, and 3; Issuance of Director's Decision Under 10 CFR § 2.206

Notice is hereby given that the Director, Office of Nuclear Reactor Regulation, has acted on a Petition for action under 10 CFR § 2.206 received from Mr. Thomas J. Saporito, Jr., on behalf of Florida Energy Consultants, Inc., dated May 27, 1994, as supplemented on July 8, 1994, for the Palo Verde Nuclear Generating Station, Unit Nos. 1, 2, and 3.

In a letter dated May 27, 1994, the Petitioner requested that the NRC (1) institute a show-cause proceeding pursuant to 10 CFR § 2.202 to modify, suspend, or revoke the operating licenses for Palo Verde; (2) issue a notice of violation against the licensee for continuing to employ The Atlantic Group (TAG) as a labor contractor at Palo Verde; (3) investigate alleged material false statements made by William F. Conway, Executive Vice President at Palo Verde, during his testimony at a Department of Labor hearing (ERA Case No. 92–ERA–030) and, in the interim, require that he be relieved of any authority over operations at Palo Verde; (4) investigate the licensee's statements in a letter of August 10, 1993, from Mr. Conway to the former NRC regional administrator, Mr. Bobby H. Faulkenberry, that Mr. Saporito gave materially false, inaccurate, and incomplete information on his application for unescorted access to Palo Verde and that, as a result, he lacks trustworthiness and reliability for access to Palo Verde; (5) investigate the circumstances surrounding the February 1994 termination of licensee employee Joseph Straub, a former radiation protection technician at Palo Verde, to determine if his employment was illegally terminated by the licensee because he engaged in ''protected activity'' during the course of his employment; (6) require the licensee to respond to a ''chilling effect'' letter regarding the circumstances surrounding Mr. Straub's termination from Palo Verde and to specify whether any measures were taken to ensure that his termination did not have a chilling effect at Palo Verde; and (7) initiate appropriate actions to require the licensee to immediately conduct eddy current testing on all steam generators at Palo Verde because the steam generator tubes were recently found to be subject to cracking.

In a letter dated July 8, 1994, the Petitioner raised six additional issues. This supplemental Petition asked the NRC to (1) institute a show-cause proceeding pursuant to 10 CFR § 2.202 for the modification, suspension, or revocation of the Palo Verde operating licenses for Units 1, 2, and 3; (2) modify the Palo Verde operating licenses to require operation at 86-percent power or less; (3) require the licensee to submit a No Significant Hazards safety analysis to justify operation of those units above 86-percent power; (4) take immediate action (e.g., by confirmatory order) to make the licensee reduce operation to 86-percent power or less; (5) require the licensee to analyze a design-basis steam generator tube rupture (SGTR) event to show that the offsite radiological consequences do not exceed a small fraction of the limits of 10 CFR Part 100; and (6) require the licensee to demonstrate that its emergency operating procedures for SGTR events are adequate and that the plant operators are sufficiently trained in emergency operating procedures.

The Director of the Office of Nuclear Reactor Regulation has determined that requests 1, 2, 3, 5, and 6 of the July 8, 1994, Petition supplement should be denied for the reasons stated in the ''Director's Decision Under 10 CFR § 2.206'' (DD–96–08), the complete text of which follows this notice and which is available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street, N.W., Washington, D.C. 20555, and at the local public document room located at the Phoenix Public Library, 1221 N. Central Avenue, Phoenix, Arizona 85004. The Petitioners' two requests for immediate action (Request 7 of the May 27, 1994 Petition and Request 4 of the July 8, 1994, Petition supplement) were denied in a letter dated July 26, 1994. The remaining requests are under consideration and will be addressed in a separate decision. A Director's Decision (DD–96–04) regarding Requests 1 through 6 in the Petition of May 27, 1994, was issued under separate cover letter on June 3, 1996.

A copy of this Decision will be filed with the Secretary for the Commission's review in accordance with 10 CFR § 2.206. As provided by the regulation, the Decision will constitute the final action of the Commission 25 days after the date of issuance of the Decision unless the Commission on its own motion institutes a review of the Decision within that time.

Dated at Rockville, Maryland, this 25th day of June 1996.

For the Nuclear Regulatory Commission.

William T. Russell,

Director, Office of Nuclear Reactor Regulation.

I. Introduction

On May 27, 1994, Florida Energy Consultants, Inc. (FEC), by and through Thomas J. Saporito, Jr. (Petitioners), submitted a Petition pursuant to 10 CFR § 2.206 to the U.S. Nuclear Regulatory Commission (NRC). The Petition requested that the NRC (1) institute a show-cause proceeding pursuant to 10 CFR § 2.202 to modify, suspend, or revoke the operating licenses of Arizona Public Service Company (licensee or APS) for Palo Verde Nuclear Generating Station (PVNGS or Palo Verde); (2) issue a notice of violation against the licensee for continuing to employ The Atlantic Group (TAG) as a labor contractor at Palo Verde; (3) investigate alleged material false statements made by William F. Conway, Executive Vice President at Palo Verde, during his testimony at a Department of Labor hearing (ERA Case No. 92–ERA–030) and, in the interim, require that he be relieved of any authority over operations at Palo Verde; (4) investigate the licensee's statements in a letter dated August 10, 1993, from Mr. Conway to the former NRC regional administrator, Mr. Bobby H. Faulkenberry, that Mr. Saporito gave materially false, inaccurate, and incomplete information on his application for unescorted access to Palo Verde and that, as a result, Mr. Saporito lacks trustworthiness and reliability for access to Palo Verde; (5) investigate the circumstances surrounding the February 1994 termination of licensee employee Joseph Straub, a former radiation protection technician at Palo Verde, to determine if his employment was illegally terminated by the licensee because he engaged in ''protected activity'' during the course of his employment; (6) require the licensee to respond to a ''chilling effect'' letter regarding the circumstances surrounding Mr. Straub's termination from Palo Verde and specify whether any measures were taken to ensure that his termination did not have a chilling effect at Palo Verde; and (7) initiate appropriate actions to require the licensee to immediately conduct eddy current testing (ECT) on all steam generators (SGs) at Palo Verde because the SG tubes were recently found to be subject to cracking.

As the bases for these requests, the Petitioners allege that (1) a show-cause proceeding is necessary (a) because the public health and safety concerns alleged are significant and (b) to permit public participation to provide NRC with new and relevant information; (2) past practices of TAG demonstrate that employees of TAG were retaliated against for having raised safety concerns while employed at Palo Verde; (3) citations to testimony from transcripts and newspaper articles (appended as exhibits to the Petition) demonstrate that Mr. Conway's testimony is not credible; (4) statements in the letter of August 10, 1993, are inaccurate and materially false and characterize Mr. Saporito as an individual lacking trustworthiness and reliability for access to Palo Verde, and that such negative characterizations have caused the nuclear industry to blacklist him from continued employment, all in retaliation for his raising safety concerns about operations at Palo Verde; thus, the Petitioners ask that these statements be rescinded; (5) an investigation into the termination of Mr. Straub is warranted in view of the fact that the licensee has engaged in similar illegal conduct in the past for which the NRC has required the licensee to pay fines; (6) Mr. Straub is entitled to reinstatement with pay and benefits pending the NRC's investigation into his termination to offset the chilling effect his termination had on the Palo Verde workforce; and (7) in addition to cooling tower problems, the stress-corrosion and cracking in the SGs is a recurring problem of which the licensee is aware and has failed to properly correct, so that the NRC should be concerned about proper maintenance of safety systems and equipment at Palo Verde.

On July 8, 1994, the Petitioners filed a supplemental Petition (Petition supplement) raising six additional issues. The Petitioners requested that the NRC (1) institute a show-cause proceeding pursuant to 10 CFR § 2.202 for the modification, suspension, or revocation of the Palo Verde operating licenses for Units 1, 2, and 3; (2) modify the Palo Verde operating licenses to require operation at 86-percent power or less; (3) require the licensee to submit a No Significant Hazards safety analysis [1] to justify operation of those units above 86-percent power; (4) take immediate action (e.g., by confirmatory order) to require the licensee to reduce operation to 86-percent power or less; (5) require the licensee to analyze a design-basis steam generator tube rupture (SGTR) event to show that the offsite radiological consequences do not exceed a small fraction of the limits of 10 CFR Part 100; and (6) require the licensee to demonstrate that its emergency operating procedures (EOPs) for steam generator (SG) tube rupture events are adequate and the plant operators are sufficiently trained in EOPs.

As bases for these requests, the Petitioners allege that (1) the licensee experienced an SGTR in the free-span area on Unit 2 on March 14, 1993; (2) during a January 1994 inspection on

[1] Section 50.91 of the Commission's regulations provides that at the time a licensee requests an amendment it must provide the NRC its analysis of the issue of no significant hazards consideration, using the standards of Section 50.92.

Unit 2, 85 axial indications were identified, the longest indication being 7.5 inches; (3) as of May 1994, 28 axial indications were found at Unit 2 and 9 axial indications were found at Unit 1 (more extensive testing will confirm the existence of circumferential crack indications in the expansion and transition areas); (4) in May 1994, SG sludge from Units 1 and 2 indicated a lead content of 4,000 to 6,000 ppm, which is unusually high, accelerates the crevice corrosion process, and is believed to be caused by a feedwater source deficiency; (5) in eight instances, the licensee failed to properly implement operational procedures during the SGTR event on March 14, 1993; (6) the licensee's failure to comply with approved procedures in the above-mentioned event is indicative of a problem plant that warrants further NRC action; (7) in four instances, the NRC is aware of additional licensee weaknesses regarding the SGTR event; (8) the licensee cannot ensure that the radiation dose limits are satisfied for applicable postulated accidents; (9) the licensee is not maintaining an adequate level of public protection in that the offsite dose limits will be exceeded during an SGTR; (10) the licensee cannot demonstrate that a Palo Verde unit can safely be shut down and depressurized to stop SG tube leakage before a loss of reactor water storage tank inventory; (11) SG tubes are an integral part of the reactor coolant boundary and tube failures could lead to containment bypass and the escape of radioactive fission products directly into the environment and, therefore, must be carefully considered by NRC and the licensee; (12) the licensee cannot demonstrate compliance with 10 CFR Part 50, Appendix A, which establishes the fundamental requirements for SG tube integrity; (13) the licensee has failed to comply with NRC recommendations under NUREG–0800 to show that in the case of an SGTR event, ''the offsite conditions and single failure do not exceed a small fraction of the limits of 10 CFR Part 100''; and (14) the licensee has posed an unacceptable risk to public health and safety by raising power on all three Palo Verde units above 86 percent, considering the severe degradation of the SG tubes.

In a letter dated July 26, 1994, I acknowledged receipt of the Petition of May 27, 1994, and the Petition supplement of July 8, 1994, and denied the Petitioners' two requests for immediate action. The Petitioners requested the initiation of actions to require the licensee to immediately conduct ECT on all SGs at Palo Verde (Request 7 of the May 27, 1994, Petition) and immediate action to cause the licensee to reduce operation to 86-percent power or less (Request 4 of the July 8, 1994, Petition supplement). Although these two requests for immediate action were denied, the concerns raised by the Petitioners regarding their requests for ECT and reduced power operation are addressed in this Decision.

The staff informed the Petitioners that the remaining requests were being evaluated under 10 CFR § 2.206 of the Commission's regulations and that a response would be forthcoming. This Decision addresses the Petitioners' concerns about ECT (Request 7 of the May 27, 1994, Petition), steam generator tube integrity, and emergency operating procedures for SGTR events and the remaining requests (Requests 1, 2, 3, 5, and 6) of the July 8, 1994, supplement. The staff has completed its review of the remaining issues in your supplemental Petition. A Director's Decision (DD–96–04) regarding Requests 1 through 6 in the Petition of May 27, 1994, was issued under separate cover letter on June 3, 1996. A discussion of the Director's Decision follows.

II. Background

The Petitioners' concerns addressed in this Decision appear to be based largely on the March 1993 SGTR event and the NRC staff findings concerning that event set forth in the NRC Augmented Inspection Team (AIT)[2] report. Palo Verde Unit 2 experienced an SGTR event in SG No. 2 on March 14, 1993. At the time, the unit was at about 98-percent power. The plant operators manually tripped the reactor, declared an Unusual Event,[3] which was subsequently upgraded to an Alert,[4] and entered the PVNGS Functional Recovery Procedure[5] to mitigate the event. The plant was cooled down and depressurized, and the event was terminated when Mode 5[6] was achieved on March 15, 1993.

During the period March 17–25, 1993, an NRC AIT conducted an inspection at PVNGS Unit 2. Overall, the AIT concluded that the response to the SGTR succeeded in bringing the unit safely to a cold-shutdown condition and limiting the release of radioactivity so that there was no threat to public health and safety. However, the AIT identified weaknesses in the licensee's implementation of emergency plan actions, including event classification, activation of the emergency response facilities, and prompt assignment of tasks to onsite personnel. Weaknesses were also found in the procedures, equipment, and training associated with responding to an SGTR event. The AIT inspection was documented in NRC Inspection Report No. 50–529/93–14, issued on April 16, 1993.

Enforcement action resulted from the AIT inspection in several areas (e.g., emergency preparedness, chemistry and radiation monitoring, and emergency operating procedures). All violations were issued as Severity Level IV.[7]

The NRC issued a confirmatory action letter[8] (CAL) to the licensee on June 4, 1993, for Unit 2. The NRC issued a safety evaluation by letter dated August 19, 1993, concluding that Unit 2 could safely resume operation for 6 months, the interval between steam generator tube inspections. This safety evaluation closed the CAL.

The NRC issued a second CAL[9] on October 4, 1993, for Unit 3 (amended on

[2] An AIT is an NRC inspection team composed of experts from the responsible NRC Regional Office augmented by personnel from NRC Headquarters and others Regions with special technical qualifications. The purpose of an AIT is to determine the causes, conditions, and circumstances relevant to an event and to communicate its findings, safety concerns, and recommendations to NRC management.

[3] The lowest level of emergency classification as delineated in 10 C.F.R Part 50, Appendix E.

[4] The second lowest level of emergency classification as delineated in 10 C.F.R. Part 50, Appendix E.

[5] PVNGS Procedures providing operators' actions for responding to design basis events.

[6] The operational mode defined as cold shutdown in plant technical specifications.

[7] *See* EA 93–119 (issued July 1, 1993) and EA 93–039 (issued April 27, 1993). At the time, violations were categorized in terms of five levels of severity. Severity Level I and II violations were of very significant regulatory concern. Severity III violations were cause for significant regulatory concern. Severity Level IV violations were less serious but were of more than minor concern. Severity Level V were of minor safety or environmental concern. General Statement of Policy and Procedure for NRC Enforcement Actions, 10 CFR Part 2, Appendix C, Section IV. Effective June 30, 1995, the NRC's Enforcement Policy, as published in the Federal Register (60 FR 34381), is set forth in NUREG–1600.

[8] This CAL set forth commitments made by the licensee to the NRC staff on June 2, 1993, regarding the SGTR event on Unit 2. In the CAL, the staff confirmed the licensee's commitment (1) to notify the NRC prior to completion of ECT on the Unit 2 SGs; (2) to include the proposed operating interval to the next SG tube inspection in its safety analysis; and (3) not to restart Unit 2 until the NRC concurs with the restart of the facility.

[9] In this CAL, the staff confirmed the licensee's commitment to (1) shut down Unit 3 for ECT inspection of both SGs; (2) continue the review of Unit 3 ECT data to identify indications that were not identified in refueling outage 3R3 by bobbin coil ECT and to provide a written summary of the review; (3) continue to implement the Unit 1 SG inspection plan (SGIP); (4) implement changes to emergency operating procedures (EOPs), operator training, and leakage monitoring; and (5) continue to operate Unit 3 to take advantage of some of the preventive measures that can be taken to reduce

November 8 and 23, 1993), confirming the commitments made by the licensee in its September 29, 1993, letter. By letter dated December 3, 1993, the licensee reported that it had completed the actions discussed in the CAL. Satisfied that the licensee had completed the conditions of the CAL, the staff closed the CAL by letter dated April 1, 1994.

The licensee voluntarily reduced power to approximately 86-percent power in the fall of 1993 to minimize steam generator degradation. The licensee evaluated and implemented several improvements to the operation of its steam generators, one of which was a reduction in the reactor coolant system hot-leg temperature. The units were all returned to 100-percent power by the fall of 1994.

Following a midcycle outage on Unit 2 and midcycle and refueling outages on Unit 3, the NRC issued a safety evaluation on June 22, 1994, which concluded that both Unit 2 and 3 could safely operate for 6 months between steam generator tube inspections. Since that time, there have been additional midcycle outages on Units 2 and 3 and a refueling outage on all three units. Eddy current inspection results and outage planning for the Units currently support the following operating intervals between inspections: Unit 1, 16 months; Unit 2, 12 months; and Unit 3, 11 months.

III. Discussion

A. Eddy Current Testing on All Steam Generators at Palo Verde

Item 7 of the Petitioners' letter of May 27, 1994, requested the NRC to require the licensee to conduct immediate ECT on all SGs at Palo Verde to ascertain the integrity and life expectancy of the SG tubes. Although, as indicated above, this request for immediate action has been denied, the Petitioners' concerns regarding ECT are addressed below.

The Petitioners assert as a basis (Petition Basis 7) for their request concerning ECT that the licensee's SGs have recently developed cracks in the free-span portion of their internal structure, that tube stress corrosion and cracking is a recurring problem in SGs, and that there is a risk the emergency cooling system will be unable to prevent the melting of the fuel because of tube ruptures.[10]

The licensee has completed at least two eddy current inspections on each of the Palo Verde units since the SGTR event in March 1993. The staff issued safety evaluations (SEs) that addressed Unit 2 and 3 operating intervals by letters dated August 19, 1993, and June 22, 1994.[11] These SEs were based on the results of the licensee's eddy current inspections of Unit 1 in October 1993, of Unit 2 in May 1993 and January 1994, and of Unit 3 in December 1993 and May 1994. In summary, the staff concluded that Units 2 and 3 could be safely operated for up to 6 months between SG eddy current inspections. The licensee conducted five of these ''minicycles'' [12] (three on Unit 2 and two on Unit 3), thereby obtaining extensive SG eddy current data, which it used to validate models used for analysis. In May 1995, the licensee submitted a report supporting a cycle length of up to 11 months on Unit 3. Unit 1 completed a 16-month operating cycle in June 1995. After meeting with the licensee, the staff approved a Unit 3 cycle length of 11 months in a meeting summary dated August 22, 1995. During a September 20, 1995, meeting with the staff, the licensee presented its submittal and arguments to support a 12-month cycle for Unit 2. The staff incorporated data from the most recent Unit 3 steam generator inspection in its evaluation of the licensee's conclusion regarding a 12-month operating cycle on Unit 2. The staff approved the 12-month operating cycle by letter dated March 5, 1996.

In summary, the licensee performed the necessary eddy current inspections, and the staff extensively reviewed and approved Palo Verde SG eddy current inspection results and continues to review additional information regarding the integrity of the SG tubes. On the basis of its review of ECT, the staff has concluded that the Petitioners' concerns regarding the need for ECT have been satisfactorily addressed by the licensee and that no further action by the NRC staff is warranted.

B. Operation Above 86–Percent Power

Requests 1, 2, 3, and 4 of the Petition supplement, in essence, request actions requiring the Palo Verde licenses to be modified to require operation at 86-percent power or less.[13]

As bases for these requests, the Petitioners assert that on March 14, 1993, Palo Verde Unit 2 had an SGTR in the free-span section between the tube supports and that in January 1994, an inspection of Palo Verde's Unit 2 SGs found 85 axial indications (longest indication, 7.5 inches) (Petition supplement, Basis 2); and that as of May 1994, 28 axial indications were found at Unit 2 and 9 axial indications found at Unit 1. The Petitioners believe that more extensive testing will confirm the existence of circumferential crack indications in the expansion-transition area (Petition supplement, Basis 3). The Petitioners also assert that in May 1994, Units 1 and 2 SG sludge indicated a lead content of 4,000–6,000 ppm, which would accelerate the crevice corrosion cracking process (Petition supplement, Basis 4). The Petitioners also stated that the operation of Palo Verde units at above 86-percent power is unacceptable due to severe degradation of the SG tubes (Petition supplement, Basis 14).

Axial and Circumferential Steam Generator Tube Indications

With regard to the Petitioners' concern about identifiable axial indications (Petition supplement Basis 2), it is correct that 85 axial indications in the free-span area (longest indication, 7.5 inches) were discovered on SG tubes at Palo Verde Unit 2 during the January 1994 inspection. However, this number was apparently based on preliminary information from the licensee's eddy current inspection during the January 1994 eddy current inspection. The licensee's report of March 8, 1994, stated that actually 330 free-span axial indications were discovered during the Unit 2 first midcycle outage: 22 in SG 1 of Unit 2 (SG 21) and 308 in SG 2 of

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outside-diameter stress corrosion cracking (ODSCC) rates.

[10] The Petitioner also mentioned cooling tower problems in this basis, stating that ''the NRC should be concerned about proper maintenance of safety systems and equipment there.'' The cooling towers at Palo Verde are not safety-related systems. If the cooling towers of a unit were incapacitated, the unit might operate less efficiently, but that would be an economic penalty, rather than a safety problem. The Petitioners did not provide any specific examples of problems with the cooling towers, though the staff is aware of the general maintenance problems the licensee has had with the cooling towers. This issue was the subject of a previous Director's Decision, Arizona Public Service Company, (Palo Verde Nuclear Generating Station, Units 1, 2, and 3, DD–92–1, 35 NRC 133, 137 (1992), which found no substantial nuclear safety concern with the condition of the cooling towers.

[11] Unit 1 was not directly addressed in the SEs because no free span axial indications were identified on Unit 1 at the time.

[12] The Palo Verde operating cycle is normally 16–18 months.

[13] The specific request for immediate action to make the licensee reduce operation to 86-percent power or less (Request 4) was denied by letter of July 26, 1994. With regard to the request (Request 3) to require the licensee to submit a No Significant Hazards safety analysis to justify operation of the units above 86-percent power, the licensee is not required by the NRC regulations to submit a no significant hazards analysis, since a TS change was not required to resume operation above 86-percent power. The staff did, however, review a no significant hazards analysis related to operation of the Units at 100-percent power with a reduced hot-leg temperature. These TS changes were submitted by the licensee on February 18, 1994, for Units 1 and 3; and on July 1, 1994, for Unit 2. The NRC staff review of these TS changes and support for operation at a power level of 100 percent is discussed at page 17, *infra.*

Unit 2 (SG 22). Although a number of axial indications were detected by the licensee, it is not the number of indications that is of a safety concern but rather the severity of the indications (i.e., severity in terms of whether the tube indication had adequate structural and leakage integrity). As noted in the Petition supplement, the longest indication was 7.5 inches long. The safety significance of this indication, as with any eddy current indication, depends not only on the length of the indication but also on the depth of the indication. To assess the safety significance and/or severity of an indication, licensees size the indications in terms of length, depth, and/or voltage.[14] However, eddy current testing methods have not been qualified for determining the depth of stress corrosion cracks. Where qualified eddy current methods do not exist, licensees may pursue alternative methods such as in situ pressure testing[15] to further confirm or assess the condition of the tube (i.e., to confirm that the tube indication could withstand the required pressure loadings; thereby demonstrating that the tube had adequate structural integrity). The licensee did select nine tubes for in situ pressure testing during the outage. The 7.5 inch long indication did not meet the licensee's screening criteria for selecting the more severe indications. The screening criteria, discussed in the NRC staff's SE of June 22, 1994, considered the length, depth, and/or voltage of the indication. All nine tubes satisfactorily passed the in situ pressure test thereby providing reasonable assurance that the tube indications had adequate structural integrity. Furthermore, all tubes with axial free span indications were plugged before Unit 2 was returned to operation.

The Petitioners also assert as of May 1994, 28 axial indications were found on Unit 2 and 9 axial indications found at Unit 1 and that more extensive testing would confirm the existence of circumferential crack indications in the expansion transition areas (Petition supplement, Basis 3). These numbers are incorrect. Neither Unit 1 nor Unit 2 was in an outage conducting eddy current examinations in May 1994. Unit 1 had no axial indications identified as of this date. The Unit 2 data is described above. Unit 3 was in an outage at this time and identified a total of 20 axial indications. Regarding the performance of more extensive testing to confirm the existence of circumferential crack indications at the expansion transition area, the licensee has performed inspections in this region. In general, the licensee's steam generator tube inspection program consists of an initial inspection sample which is expanded, if necessary, based on the initial inspection sample results. The licensee has been examining the expansion transition locations with a motorized rotating pancake coil (MRPC) probe since, at least, 1993. These examinations permit the licensee to detect circumferential crack indications. In its SEs and meeting summaries, the NRC staff has reviewed the licensee's results from its MRPC inspections and found them acceptable.[16] To date, Palo Verde Units 2 and 3 have each exhibited a small number of circumferential crack indications per Unit. Unit 1 has exhibited the most extensive circumferential cracking both in terms of number of indications and the severity of the indications when compared to Units 2 and 3. Nonetheless, the staff concluded in a meeting summary dated October 19, 1994, that operating Unit 1 to the end of the operating cycle (April 1995) did not pose an undue risk to public health and safety in view of (1) the absence of detectable axial free-span cracks during the previous refueling outage inspection; (2) the improved secondary water chemistry performance at Palo Verde; (3) the reduced hot-leg temperature, which is expected to reduce crack growth rates; and (4) the implementation of enhanced MRPC inspection techniques at the expansion transition locations. The licensee will continue to perform extensive SG inspections at the end of each operating cycle to ensure continued safe operation of SGs.

Lead Content in Steam Generator Tube Sludge

The Petitioners assert without providing any supporting basis that the SG sludge of Units 1 and 2 has a lead content of 4,000–6,000 ppm (Petition supplement, Basis 4). The licensee performed sludge analyses during two consecutive Unit 1 outages. The data, which were reported in a letter from the licensee dated November 2, 1993, indicate a lead content of 78 ppm (from Unit 1, Refueling 3) and 98 ppm (Unit 1, Refueling 4).[17] Sludge samples were obtained from both Unit 2 SGs after the March 1993 SGTR event. The data were documented in the licensee's report, ''Equipment Root Cause of Failure.'' Both the licensee and outside contractors analyzed the samples; all analyses indicated a lead content of 100 ppm or less.

The NRC staff conducted two Palo Verde chemistry inspections (Inspection Reports 94–15 and 94–27 on Units 50–528/50–529/50–530). The staff reviewed films and sludge for their lead content, and the data were consistent with the licensee's reports. Inspection Report 50–528/50–529/50–530/94–15 specifically referred to the inspector's determination to note ''whether lead was detected, because of recent work which indicated it may have a deleterious effect.'' In referring to examinations of the burst region[18] of pulled tubes, the report stated that insignificant levels of lead were found in the sludge and in the films examined.

Inspection Report 50–528/50–529/50–530/94–15 also reviewed the licensee's secondary water chemistry control program.[19] The NRC inspection team found that the program requirements had fully conformed to the EPRI guidelines throughout Palo Verde's operating history with respect to chemical parameters, analytical frequency, limits for critical parameters, and required actions when critical parameters were exceeded. In summary, the Petitioners' assertions regarding lead content have not been substantiated and do not agree with available data. The licensee has verified[20] that lead content in both Units 1 and 2 SGs is 100 ppm or less, not 4,000–6,000 ppm as asserted by the Petitioners. Additionally, NRC Inspection Reports 94–15 and 94–27 on Units 50–528/50–529/50–530 have not

[14] Voltage is electrical force or potential difference. Voltage measurements can be used to estimate the severity of an indication.

[15] In situ pressure tests were conducted to determine whether the tubes could withstand the pressure loading specified in NRC Regulatory Guide 1.121 (i.e., whether the SG tubes have adequate structural integrity).

[16] The Staff's reviews are documented in SEs dated August 19, 1993, and June 22, 1994, and also in meeting summaries dated August 22, 1995, March 22, 1994, October 19, 1994, August 22, 1995, and September 20, 1995.

[17] During the Unit 2 midcycle outage in early 1994, the SGs were chemically cleaned before sludge lancing; therefore, the composition of the sludge was not tested.

[18] Burst region refers to the section of the crack in a pulled tube that is exposed as the result of a burst or rupture due to an applied pressure either during plant operation or laboratory testing.

[19] The NRC inspection team compared Electric Power Research Institue (EPRI) NP–6239, ''PWR Secondary Water Chemistry Guidelines,'' Revisions 1 through 2, and EPRI TR–101230, ''Interim PWR Secondary Water Chemistry Recommendations for IGA/IGSCC Control,'' with the licensee's secondary water chemistry control program for PVNGS.

[20] PVNGS performed its own inspections and also utilized contractors, ABB-Combustion Engineering (ABB-CE) and Babcock and Wilcox Nuclear Technologies (BWNT), to perform metallurgical examinations. The inspections revealed minor quantities of lead in surface deposits and films. See NRC Inspection Report 50–528/50–529/50–530/94–15, dated June 23, 1994.

revealed any information about elevated lead content.

Steam Generator Tube Degradation and Operation at a Reduced Power Level

The Petitioners also assert that the operation of Palo Verde units at above 86-percent power is unacceptable due to severe degradation of SG tubes (Petition supplement, Basis 14). In December 1993, the licensee volunteered to reduce power in all three units to approximately 86 percent as an interim measure to curtail steam generator degradation. The primary purpose of this administrative power limit was to operate with a lower reactor coolant system hot-leg temperature in order to reduce tube degradation. This specific power level had been selected because it provided for a T_{hot} that approximated the value that would be implemented if the licensee's proposed TS changes for operating at 100% power with a reduced T_{hot} were approved by the NRC. Additionally, the licensee's thermal-hydraulic analysis indicated that, at this reduced power level, the potential for freespan tube degradation from corrosion is reduced. The licensee took this action voluntarily to minimize further degradation of the SGs until corrective, mitigative, and preventive actions could be implemented to reduce steam generator tube degradation.

On June 7, 1994, the NRC issued a TS change for Units 1 and 3 that permitted the licensee to operate at full power with a lower T_{hot} temperature.[21] The Unit 2 TS change was reviewed separately because the licensee was continuing to perform analyses arising from the SG tube plugging in Unit 2. The staff issued this TS change on August 12, 1994.[22] These TS changes permitted operation at a power level of 100 percent as did the staff's post-March 1993 SGTR SEs dated August 19, 1993, and June 22, 1994, regarding the length of operating cycles of the Palo Verde units. Furthermore, as stated above, the staff did not impose any power restrictions or limits on the licensee.

In summary, the Petitioners' concerns regarding operation of the Palo Verde units above 86-percent power (including bases relating to the March 1993 SGTR event, identification of axial and circumferential steam generator tube indications, alleged elevated lead contents in steam generator sludge) have been satisfactorily addressed, and do not warrant any further action by the NRC staff.

C. Need To Reanalyze the Design-Basis SGTR Event

Request 5 (of the Petition supplement) is that the NRC require the licensee to analyze a design-basis SGTR event to show that the offsite radiological consequences do not exceed a small fraction of the limits of 10 CFR Part 100. The staff requires an analysis such as this to be completed for all pressurized-water reactors (PWRs) and documented in a final safety analysis report (FSAR) before plant operation. The licensee complied with this requirement.[23]

The Petitioners assert in the basis (Petition supplement, Bases 8, 9, 10, 11 and 13) that the licensee cannot ensure the dose limits are satisfied for applicable postulated SGTR events; the offsite dose limits would be exceeded during an SGTR event and adequate protection to the public would not be maintained; the licensee cannot demonstrate that the plant can be safely shut down and depressurized to stop SG tube leakage before reactor water storage tank inventory is lost; the NRC and the licensee must carefully consider SGTR; and ''the licensee has failed to comply with NRC requirements under NUREG–0800 insofar as the licensee is required to analyze the consequences of a design basis SGTR event to show that the offsite conditions and single failure do not exceed a small fraction of limits of 10 CFR Part 100.''

The AIT report documents findings regarding the Unit 2 SGTR event of March 1993. The report stated that the plant was safely brought to cold shutdown and no radioactivity was released off site. Additionally, the staff's SE, dated August 19, 1993, assessed a single SGTR event and single and multiple tube ruptures induced by a major secondary-side rapid depressurization and concluded that the radiological consequences were within applicable limits.[24] Finally, in a memorandum dated January 26, 1996, the staff performed a confirmatory review of the licensee's updated SGTR event analysis, submitted with Revision 6 to the FSAR (March 10, 1994), and concluded that the results are acceptable. The Petitioners also assert in the basis (Petition supplement, Basis 12) that the licensee cannot demonstrate compliance with certain criteria of Appendix A to 10 CFR Part 50,[25] which establishes the fundamental requirements for steam generator tube integrity. However, the Petitioners have failed to provide any details or support for this assertion.

In summary, on the basis of the NRC staff's review of the licensee's design-basis SGTR event and more recent confirmatory review, the staff has concluded that the Petitioners have not presented a basis for further NRC action.

D. Adequacy of Training and Procedures for an SGTR Event

Regarding Request 6 of the Petition supplement, that the NRC require the licensee to demonstrate that its emergency operating procedures (EOPs) for SGTR events are adequate and the plant operators are sufficiently trained in EOPs, the staff has already taken sufficient action. The Petitioners allege (Petition supplement, Bases 5, 6, and 7, respectively) that the licensee failed to properly implement operational procedures regarding the SGTR event of March 14, 1993, citing eight instances in Basis 5 [26]; that the licensee's failure to comply with approved procedures in this event is indicative of a problem plant that warrants further NRC attention (Basis 6); and that the NRC is aware of additional licensee weaknesses regarding the SGTR event, citing four instances in Basis 7.[27] These bases

[21] Noticed in the Federal Register on June 22, 1994 (59 Fed. Reg. 32240).

[22] Noticed in the Federal Register on August 31, 1994 (59 Fed. Reg. 45038).

[23] Updated Final Safety Analysis Report (UFSAR) Section 15.6.3.1.3.2 describes the radiological consequences of an SGTR, and the results are shown in UFSAR Table 15.6.3–5. The staff initially reviewed PVNGS's UFSAR in November 1981.

[24] In 10 CFR Part 100, acceptance criteria are specified for the dose analyzed during initial plant licensing at the exclusion area boundary (EAB) and low population zone (LPZ) for design-basis accidents. The dose in 2 hours at the EAB is not to exceed 25 rem to the whole body or 300 rem to the thyroid. The dose in 30 days at the boundary of the LPZ is not to exceed 25 rem to the whole body or 300 rem to the thyroid. The staff reviewed the licensee's Unit 2 steam generator tube rupture analysis, submitted by letter dated July 18, 1993, and concluded that the methods used by the licensee were acceptable. See the NRC staff's safety evaluation dated August 19, 1993.

The Petitioners assert that the licensee has failed to comply with NUREG–0800 requirements regarding consequences of a design basis SGTR event. However, NUREG–0800 does not set forth requirements; rather it sets forth acceptable approaches to satisfying NRC requirements.

[25] The Petitioners reference portions of General Design Criteria (GDC) 14, 15, 30, and 31 of Appendix A to 10 CFR Part 50.

[26] The Petitioners assert (Petition supplement, Basis 5) that the licensee (a) failed to classify the event in accordance with the EOPs, (b) failed to actuate the Emergency Operations Facility for the 1-hour time, (c) failed to activate the Emergency Response Data System, (d) violated 10 CFR § 50.72 requirements, activation of the Emergency Response Data System, (e) failed to take prompt corrective actions to repair the condenser vacuum pump exhaust radiation monitor, (f) failed to obtain required approvals for alarm setpoint change on waste gas area combined ventilation exhaust monitor, (g) failed to fully implement an alarm response procedure and, (h) failed to check the owner-controlled area.

[27] The Petitioners assert (Petition supplement, Basis 7) that the licensee's (a) alert and alarm setpoints for condenser vacuum pump exhaust and main steam line radiation monitor limits appear to be based on offsite dose limits rather on an SGTR

Continued

largely concern areas the staff reviewed after the SGTR event on March 14, 1993. Specifically, the Petitioners repeated several of the procedural and operator weaknesses that were described and evaluated in the staff's AIT report (Inspection Report 50–529/93–14, dated April 16, 1993).[28] Specifically, the AIT report stated that the use of a diagnostic logic tree caused the operators to misdiagnose the SGTR event twice and subsequently enter a Functional Recovery Procedure, contributing substantially to the delay in isolating the faulted steam generator. The staff concluded in its safety evaluation of August 19, 1993, that the licensee's modifications to the EOPs and the subsequent operator training provide sufficient enhancement for both diagnosis and mitigation of various SGTR scenarios.

Additionally, the licensee recently revised its EOPs to make them consistent with Combustion Engineering Owners Group (CEOG) guidance (CEN 0152, Rev. 3 [29]). NRC Inspection Report 50–528/50–529/50–530/95–12, dated July 27, 1995, documents the staff's observations on the ''high intensity team'' training conducted for each crew in preparation for implementing the EOPs. In the inspection report, the staff stated that the EOPs enhanced crew performance and allowed for greater flexibility in responding to events. As an example, during the simulator-based SGTR scenario, the crew was able to isolate the faulted SG within 14 minutes of the start of the event. In contrast, during the March 1993 Unit 2 SGTR event, operators took about 3 hours to isolate the faulted SG, partly because of restrictions in the EOPs in use at the time. The staff will further evaluate the effectiveness of EOPs during future licensed operator examinations.

On the basis of its review of the Petitioners' request that the licensee demonstrate that its EOPs for SGTR events are adequate and that plant operators are sufficiently trained in EOPs, the staff has concluded that the Petitioners have not presented a basis for further NRC action.

III. Conclusion

The institution of proceedings in response to a request pursuant to Section 2.206 is appropriate only when substantial health or safety issues have been raised. See Consolidated Edison Co. of New York (Indian Point, Units 1, 2, and 3), CLI–75–8, 2 NRC 173, 176 (1975), and Washington Public Power Supply System (WPPSS Nuclear Project No. 2), DD–84–7, 19 NRC 899, 923 (1984). This standard has been applied to the concerns raised by the Petitioners to determine whether the actions requested by the Petitioners are warranted. With regard to the specific requests made by the Petitioners discussed herein, the NRC staff finds no basis for taking additional actions beyond those described above. Accordingly, the Petitioners' requests for additional actions pursuant to Section 2.206, specifically Requests 1, 2, 3, 5, and 6 submitted in the Petitioners' supplement dated July 8, 1994, are denied. Accordingly, no action pursuant to Section 2.206 is being taken in this matter.

A copy of this Decision will be filed with the Secretary of the Commission for Commission review in accordance with 10 CFR § 2.206(c) of the Commission's regulations. As provided by this regulation, the Decision will constitute the final action of the Commission 25 days after issuance, unless the Commission, on its own motion, institutes a review of the Decision within that time.

Dated at Rockville, Maryland, this 25th day of June 1996.

For the Nuclear Regulatory Commission.

William T. Russell,

Director, Office of Nuclear Reactor Regulation.

[FR Doc. 96–16878 Filed 7–1–96; 8:45 am]

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event, (b) simulator alarms occur within 2–3 minutes of an SGTR event, contrary to control room indications, (c) plant staff failed to fully respond to assembly notification, (d) plant staff failed to perform a formal evaluation of the safety significance of an abnormal crack growth in the Unit 2 SG.

[28] The licensee addressed the issues raised in the AIT report by implementing the necessary procedural changes and providing training. For example, with regard to the AIT finding (summarized by the Petitioners) regarding differences between alarm response on the simulator and in the control room, the staff's safety evaluation of August 19, 1993, stated that ''the simulator has been modified to more realistically model the plant, particularly the response of the radiation monitoring system to an SGTR.''

[29] A letter from the NRC to Combustion Engineering dated August 2, 1988, stated that, ''pending NRC final review and approval, CE facilities may base their plant-specific emergency operating procedures on Revision 3 of CEN–152. Should future NRC review reveal modifications to Revision 3 to be necessary, CE facilities would be expected to update their procedures to reflect the identified changes. Schedules for such changes should be based on perceived safety significance of the changes.'' The objective of the CEN–152 report is to describe the CEOG emergency procedure guidelines system. The report contains the methodology used to develop and validate the licensee's emergency procedure guidelines and information on the implementation of guidelines.

POSTAL SERVICE

Specifications for Postal Security Devices and Indicia (Postmarks)

AGENCY: Postal Service.

ACTION: Notice of proposed specifications with request for comments.

SUMMARY: Historically, postage meters have been mechanical and electromechanical devices that (1) maintain through mechanical or electronic ''registers'' (postal security devices) an account of all postage printed and the remaining balance of prepaid postage, and (2) print postage postmarks (indicia) that are accepted by the Postal Service as evidence of the prepayment of postage. Two proposed specifications have been developed on these subjects, and are entitled ''Information Based Indicia Program (IBIP) PSD Specification'' and ''Information Based Indicia Program (IBIP) Indicia Specification.'' The U.S. Postal Service is seeking comments on these specifications.

The Postal Service also seeks comments on intellectual property issues raised by the specifications if adopted in present form. If an intellectual property issue includes patents or patent applications covering any implementations of the specifications, the comment should include a listing of such patents and applications and the license terms available for such patents and applications.

DATES: Comments on the two specifications must be received on or before September 30, 1996. Comments addressing intellectual property issues must be received on or before July 15, 1996. A general meeting on this subject is being planned for mid-July in Washington, DC. All persons who have expressed an interest in the proposed specifications will be invited to attend the meeting. This meeting will focus solely on technical aspects of the two specifications. Interested parties may submit questions by July 1, 1996 which will be considered for incorporation into the meeting presentations.

ADDRESSES: Copies of the Indicium and Postal Security Device Specifications may be obtained from: Terry Goss, United States Postal Service, 475 L'Enfant Plaza SW, Room 8430, Washington, DC 20260–6807. Mail or deliver written comments to: Manager, Retail Systems and Equipment, United States Postal Service, 475 L'Enfant Plaza SW, Room 8430, Washington, DC 20260–6807. Copies of all written comments may be inspected and