for certain drugs which contain ephedrine. Persons who previously were not required to keep records or make reports regarding sales of these products now must do so.

(5) 10,300 annual respondents at .17 hours per response. 10,000 recordkeepers at 100 annual hours per recordkeeper.

(6) 3,502 total annual reporting hours. 1,000,000 total annual recordkeeping hours. Recordkeeping retention period: 4 years. 1,003,451 Total Annual Burden.

(7) Not applicable under Section 3504(h) of Public Law 96–511.

Public comment on this item is encouraged.

Dated: September 8, 1994.

Robert B. Briggs,

Department Clearance Officer, United States Department of Justice.

[FR Doc. 94–22651 Filed 9–13–94; 8:45 am]

BILLING CODE 4410–09–M

Information Collections Under Review

The Office of Management and Budget (OMB) had been sent the following collection(s) of information proposals for review under the provisions of the Paperwork Reduction Act (44 U.S.C. chapter 35) and the Paperwork Reduction Reauthorization Act since the last list was published. Entries are grouped into submission categories, with each entry containing the following information:

(1) the title of the form/collection;
(2) the agency form number, if any, and the applicable component of the Department sponsoring the collection;
(3) how often the form must be filled out or the information is collected;
(4) who will be asked or required to respond, as well as a brief abstract;
(5) an estimate of the total number of respondents and the amount of time estimated for an average respondent to respond;
(6) an estimate of the total public burden (in hours) associated with the collection; and,
(7) an indication as to whether Section 3504(h) of Public Law 96–511 applies.

Comments and/or suggestions regarding the item(s) contained in this notice, especially regarding the estimated public burden and associated response time, should be directed to the OMB reviewer, Mr. Jeff Hill on (202) 395–7340 and to the Department of Justice's Clearance Officer, Mr. Robert B. Briggs, on (202) 514–4319. If you anticipate commenting on a form/collection, but find that time to prepare such comments will prevent you from prompt submission, you should notify the OMB reviewer and the Department of Justice Clearance Officer of your intent as soon as possible. Written comments regarding the burden estimate or any other aspect of the collection may be submitted to Office of Information and Regulatory Affairs, Office of Management and Budget, Washington, DC 20503, and to Mr. Robert B. Briggs, Department of Justice Clearance Officer, Systems Policy Staff/ Information Resources Management/ Justice Management Division Suite 850, WCTR, Washington, DC 20530.

New Collection

(1) Application for Registration under Domestic Chemical Diversion Control Act of 1993 (DEA Form 510) Renewal Application for Registration under Domestic Chemical Diversion Control Act of 1993 (DEA Form 510a)

(2) DEA Forms 510 and 510a. Drug Enforcement Administration.

(3) On Occasion. DEA Form 501, New Applicant. Annually. DEA Form 510a, Renewal.

(4) Individuals or households, Businesses or other for-profit, Small businesses or organizations. The Domestic Chemical Diversion Control Act requires that distributors, importers and exporters of listed chemicals which are being diverted in the United States for the production of illicit drugs must register with the DEA. Registration provides a system to aid in the tracking of the distribution of List I chemicals.

(5) 11,500 annual respondents at .5 hours per response.

(6) 5,750 annual burden hours.

(7) Not applicable under Section 3504(h) of Public Law 96–511.

Public comment on this item is encouraged.

Dated: September 8, 1994.

Robert B. Briggs,

Department Clearance Officer, United States Department of Justice.

[FR Doc. 94–22650 Filed 9–13–94; 8:45 am]

BILLING CODE 4410–09–M

Drug Enforcement Administration

Importer of Controlled Substances; Notice of Registration

By Notice dated May 6, 1994, and published in the **Federal Register** on May 13, 1994, (59 FR 25126), Sanofi Winthrop L.P., DBA Sanofi Winthrop Pharmaceutical, 200 East Oakton Street, Des Plaines, Illinois 60018, made application to the Drug Enforcement Administration to be registered as an importer of the basic classes of controlled substances listed below:

Drug	Schedule
Codeine (9050)	II
Hydromorphone (9150)	II
Meperidine (9230)	II
Morphine (9300)	II

Comments were received and a registered importer did file a written request for a hearing with respect to the registration of Sanofi Winthrop L.P., the firm subsequently withdrew its request for a hearing on July 22, 1994, because it intends to use the import registration to allow its Distribution Center to re-import controlled substances not acceptable to foreign customers. Therefore, pursuant to Section 1008(a) of the Controlled Substances Import and Export Act and in accordance with Title 21, Code of Federal Regulations, § 1311.42, the above firm is granted registration as an importer of the basic classes of controlled substances listed above.

Dated: September 6, 1994.

Gene R. Haislip,

Deputy Assistant Administrator, Office of Diversion Control, Drug Enforcement Administration.

[FR Doc. 94–22717 Filed 9–13–94; 8:45 am]

BILLING CODE 4410–09–M

DEPARTMENT OF LABOR

Employment and Training Administration Notice of Attestations Filed by Facilities Using Nonimmigrant Aliens as Registered Nurses

AGENCY: Employment and Training Administration, Labor.

ACTION: Notice.

SUMMARY: The Department of Labor (DOL) is publishing, for public information, a list of the following health care facilities that have submitted attestations (Form ETA 9029 and explanatory statements) to one of four Regional Offices of DOL (Boston, Chicago, Dallas and Seattle) for the purpose of employing nonimmigrant alien nurses. A decision has been made on the these organizations' attestations and they are on file with DOL.

ADDRESSES: Anyone interested in inspecting or reviewing the employer's attestation may do so at the employer's place of business.

Attestations and short supporting explanatory statements are also available for inspection in the U.S. Employment Service, Employment and Training Administration, Department of Labor, Room N–4456, 200 Constitution Avenue, N.W., Washington, D.C. 20210. Any complaints regarding a particular attestation or a facility's activities under that attestation, shall be filed with a local office of the Wage and Hour Division of the Employment Standards Administration, Department of Labor. The address of such offices are found in many local telephone directories, or may be obtained by writing to the Wage and Hour Division, Employment Standards Administration, Department of Labor, Room S–3502, 200 Constitution Avenue, N.W., Washington, D.C. 20210.

FOR FURTHER INFORMATION CONTACT:

Regarding the Attestation Process: Chief, Division of Foreign Labor Certifications, U.S. Employment Service. Telephone: 202–219–5263 (this is not a toll-free number).

Regarding the Complaint Process: Questions regarding the complaint process for the H–1A nurse attestation program will be made to the Chief, Farm Labor Program, Wage and Hour Division. Telephone: 202–219–7605 (this is not a toll-free number).

SUPPLEMENTARY INFORMATION: The Immigration and Nationality Act requires that a health care facility seeking to use nonimmigrant aliens as registered nurses first attest to the Department of Labor (DOL) that it is taking significant steps to develop, recruit and retain United States (U.S.) workers in the nursing profession. The law also requires that these foreign nurses will not adversely affect U.S. nurses and that the foreign nurses will be treated fairly. The facility's attestation must be on file with DOL before the Immigration and Naturalization Service will consider the facility's H–1A visa petitions for bringing nonimmigrant registered nurses to the United States. 26 U.S.C. 1101(a)(15)(H)(i)(a) and 1181(m). The regulations implementing the nursing attestation program are at 20 CFR Parts 655, Subpart D, and 29 CFR Part 504, (January 6, 1994). The Employment and Training Administration, pursuant to 20 CFR 655.310(c), is publishing the following list of facilities which have submitted attestations which have been accepted for filing and those which have been rejected.

The list of facilities is published so that U.S. registered nurses, and other persons and organizations can be aware of health care facilities that have requested foreign nurses for their staff. If U.S. registered nurses or other persons wish to examine the attestation (on Form ETA 9029) and the supporting documentation, the facility is required to make the attestation and documentation available. Telephone numbers of the facilities chief executive officer also are listed to aid public inquiries. In addition, attestations and explanatory statements (but not the full supporting documentation) are available for inspection at the address for the Employment and Training Administration set forth in the **ADDRESSES** section of this notice.

If a person wishes to file a complaint regarding a particular attestation or a facility's activities under the attestation, such complaint must be filed at the address for the Wage and Hour Division of the Employment Standards Administration set forth in the **ADDRESSES** section of this notice.

Signed at Washington, D.C., this 7th day of September 1994.

John M. Robinson,

Deputy Assistant Secretary, Employment and Training Administration.

DIVISION OF FOREIGN LABOR CERTIFICATIONS, HEALTH CARE FACILITY ATTESTATIONS

[FORM ETA–9029]

CEO-Name/Facility Name/Address	State	Action date
ETA REGION 1 **07/04/94 TO 07/10/94**		
David Potvin, Bethany Health Care Center, Inc., 97 Bethany Road, Framingham, MA 01701, 508–872–6750 ETA CONTROL NUMBER—1/212623 ACTION—ACCEPTED	MA	07/06/94
Neal M. Elliott, Greenery—Beverly, 40 Heather St., Beverly, MA 01915, 508–927–6620 ETA CONTROL NUMBER—1/212733 ACTION—ACCEPTED	MA	07/07/94
Neal M. Elliott, Greenery—Boston, 99 Chestnut Hill Ave., Boston, MA 02135, 787–3390 ETA CONTROL NUMBER—1/212731 ACTION—ACCEPTED	MA	07/07/94
Neal M. Elliott, Greenery—North Andover, 75 Park Street, N. Andover, MA 01845, 508–685–3372 ETA CONTROL NUMBER—1/212725 ACTION—ACCEPTED	MA	07/07/94
Neal M. Elliott, Greenery—Waltham, 775 Trapelo Road, Waltham, MA 02554, 895–7000 ETA CONTROL NUMBER—1/212726 ACTION—ACCEPTED	MA	07/07/94
Neal M. Elliott, Greenery—Worcester, 59 Acton St., Worcester, MA 01604, 508–791–3147 ETA CONTROL NUMBER—1/212724 ACTION—ACCEPTED	MA	07/07/94
Neal M. Elliott, Greenery Rehab & Skilled Nursing, 89 Lewis Bay Road, Hyannis, MA 02601, 508–775–7601 ETA CONTROL NUMBER—1/212728 ACTION—ACCEPTED	MA	07/07/94
Neal M. Elliott, Greenery Rehabilitation Center, P.O. Box 1330, Isaac Street, Middleboro, MA 02346, 508–947–9295 ETA CONTROL NUMBER—1/212727 ACTION—ACCEPTED	MA	07/07/94
Gerald L. MacDonald, Mediplex Skilled Nursing & Rehab., 910 Saratoga St., East Boston, MA 02128, 569–1157 ETA CONTROL NUMBER—1/212701 ACTION—ACCEPTED	MA	07/06/94
Nancy Hsu, South Cove Manor Nursing Home, 120 Shawmut Avenue, Boston, MA 02118, 423–0590 ETA CONTROL NUMBER—1/212552 ACTION—ACCEPTED	MA	07/06/94
Jeanne V. Sanders, Golden View Health Center Corp., 19 NH Route 104, Meredith, NH 03253, 603–279–8111 ETA CONTROL NUMBER—1/212884 ACTION—ACCEPTED	NH	07/06/94
Kristine Graff, Bristol Manor Health Care Center, 96 Parkway, Rochelle Park, NJ 07662, 201–845–0099 ETA CONTROL NUMBER—1/212550 ACTION—ACCEPTED	NJ	07/05/94
Natalie Zanetich Fatigati, Hamilton Park Health Care Center, 525–535 Monmouth St., Jersey City, NJ 07302, 201–653–8800. ETA CONTROL NUMBER—1/212554 ACTION—ACCEPTED	NJ	07/06/94
Gloria F. Estabillo, Philippino Placement Agency, Inc., 880 Bergen Avenue, Jersey City, NJ 07306, 201–983–0245 ETA CONTROL NUMBER—1/212553 ACTION—ACCEPTED	NJ	07/06/94
Joan V. Tomczyk, Beach Terrace Care Center, Inc., 640 W. Broadway, Long Beach, NY 11561, 516–431–4400 ETA CONTROL NUMBER—1/212513 ACTION—ACCEPTED	NY	07/05/94
Debra A. Sabato, Cedar Manor Nursing Home, P.O. Box 928, Cedar Lane, Ossining, NY 10562, 914–762–1600	NY	07/06/94

DIVISION OF FOREIGN LABOR CERTIFICATIONS, HEALTH CARE FACILITY ATTESTATIONS—Continued
[FORM ETA-9029]

CEO-Name/Facility Name/Address	State	Action date
ETA CONTROL NUMBER—1/212887 ACTION—ACCEPTED		
John C. Federspiel, Hudson Valley Hospital Center, 1980 Crompond Road, Peekskill, NY 10566, 914–734–3571	NY	07/05/94
ETA CONTROL NUMBER—1/212514 ACTION—ACCEPTED		
William Tan, St. Agnes Hospital, 305 North Street, White Plains, NY 10605 914–681–4507	NY	07/06/94
ETA CONTROL NUMBER—1/212700 ACTION—ACCEPTED		
ETA REGION 1 07/11/94 TO 07/17/94		
Margaret K. Degnan, Moris Hills Multicare Center, 77 Madison Avenue, Morristown, NJ 07960, 201–540–9800	NJ	07/11/94
ETA CONTROL NUMBER—1/213002 ACTION—ACCEPTED		
Abraham Schlafrig, Meadow Park Nursing Home, 78–10 164th St., Flushing, NY 11366, 718–591–8300	NY	07/14/94
ETA CONTROL NUMBER—1/213131 ACTION—ACCEPTED		
Patricia Lambert, New York State Kingsboro Psych. Ctr, 681 Clarkson Avenue, Brooklyn, NY 11203–2199, 718–221–7100.	NY	07/12/94
ETA CONTROL NUMBER—1/213047 ACTION—ACCEPTED		
ETA REGION 1 07/18/94 TO 07/24/94		
Jonathan M. Metsch, Greenville Hospital, 1825 Kennedy Boulevard, Jersey City, NJ 07305, 201–547–6100	NJ	07/18/94
ETA CONTROL NUMBER—1/213200 ACTION—ACCEPTED		
Edward Mr. Einhorn, Hospitality Care Center, 300 Broadway, Newark, NJ 07104, 201–484–4222	NJ	07/18/94
ETA CONTROL NUMBER—1/213203 ACTION—ACCEPTED		
Charles P. Berkowitz, Jewish Home & Rehabilitation Ctr., 198 Stevens Avenue, Jersey City, NJ 07305, 201–451–9000	NJ	07/18/94
ETA CONTROL NUMBER—1/213168 ACTION—ACCEPTED		
Alvin J. Conway, Catholic Med. Ctr./Brooklyn & Queens, 88–25 153rd Street, Jamaica, NY 11432, 718–630–6800	NY	07/18/94
ETA CONTROL NUMBER—1/213201 ACTION—ACCEPTED		
ETA REGION 1 07/25/94 TO 07/31/94		
Maurice I. May, Hebrew Rehab Ctr for the Aged, 1200 Centre Street, Boston, MA 02131, 617–325–8000	MA	07/26/94
ETA CONTROL NUMBER—1/213425 ACTION—ACCEPTED		
Michele B. Anderson, ARO Community Services, Inc., 11 Northeastern Blvd., Nashua, NH 03062–3139, 603–598–9800.	NH	07/26/94
ETA CONTROL NUMBER—1/213388 ACTION—REJECTED		
Raymond C. Lemire, Epsom Manor, Inc., Junction Route 4 on 28 RR2, Box 107, Epsom, NH 03234, 603–736–4772	NH	07/25/94
ETA CONTROL NUMBER—1/213303 ACTION—ACCEPTED		
Benjamin F. Miller, Delaire Nursing & Convalescent Ctr, 400 W. Stimpson Ave., Linden, NJ 07036, 908–862–3399	NJ	07/26/94
ETA CONTROL NUMBER—1/213372 ACTION—ACCEPTED		
Lowell Fein, Eagle Rock Convalescent Center, T/A West Caldwell Care Center 165, Fairfield Avenue, West Caldwell, NJ 07006, 201–226–1100.	NJ	07/25/94
ETA CONTROL NUMBER—1/213297 ACTION—ACCEPTED		
Michael J. McDonough, Hospital Center at Orange (The), 188 South Essex Avenue, Orange, NJ 07050, 201–266–2269.	NJ	07/29/94
ETA CONTROL NUMBER—1/213498 ACTION—ACCEPTED		
John P. McGee, JFK Health Systems, Inc., 65 James St., Edison, NJ 08818–3059, 908–321–7170	NJ	07/25/94
ETA CONTROL NUMBER—1/213387 ACTION—ACCEPTED		
Harvey Holzberg, Robert Wood Johnson Univ. Hospital, 1 RobertWood Jonson Place, New Brunswick, NJ 08901, 908–828–3000.	NJ	07/26/94
ETA CONTROL NUMBER—1/213426 ACTION—ACCEPTED		
James Davis, Amsterdam Nursing Home, 1060 Amsterdam Avenue, New York, NY 10025, 212–678–2600	NY	07/25/94
ETA CONTROL NUMBER—1/213296 ACTION—ACCEPTED		
A. Dixon, Bayview Correctional Facility, 550 West 20th Street, New York, NY 10011, 212–255–7590	NY	07/26/94
ETA CONTROL NUMBER—1/213371 ACTION—ACCEPTED		
Raquel Ayala, Central Bronx Hospital, Immigration Unit, 125 Worth Street, New York, NY 10013, 212–788–3485	NY	07/25/94
ETA CONTROL NUMBER—1/213293 ACTION—ACCEPTED		
George Adams, Lutheran Medical Center, 150 55th Street, Brooklyn, NY 11220, 718–630–7000	NY	07/26/94
ETA CONTROL NUMBER—1/213374 ACTION—ACCEPTED		
Raquel Ayala, New York City Health & Hospitals, Immigration Unit, 125 Worth St., New York, NY 10013, 212–788–3500.	NY	07/25/94
ETA CONTROL NUMBER—1/213291 ACTION—ACCEPTED		
George H. McCoy, St. Croix Hospital, 4007 Est. Diamond Ruby, St. Croix, VI 00820–4421, 809–778–6311	VI	07/29/94
ETA CONTROL NUMBER—1/213499 ACTION—ACCEPTED		
ETA REGION 1 08/01/94 TO 08/07/94		
Joseph Barrick, Riverdale Gardens Inc., 42 Prospect Avenue, West Springfield, MA 01089, 413–733–3151	MA	08/01/94
ETA CONTROL NUMBER—1/213521 ACTION—ACCEPTED		
Lawrence N. Stein, King James Care Center, 1501 State Highway 33, Hamilton Square, NJ 08690, 609–586–1114	NJ	08/02/94

DIVISION OF FOREIGN LABOR CERTIFICATIONS, HEALTH CARE FACILITY ATTESTATIONS—Continued
[FORM ETA-9029]

CEO-Name/Facility Name/Address	State	Action date
ETA CONTROL NUMBER—1/213582 ACTION—ACCEPTED		
C. Beth Kelly, Lakewood Nursing Center, 285 River Avenue, Lakewood, NJ 08701, 908-363-0400	NJ	08/01/94
ETA CONTROL NUMBER—1/213524 ACTION—REJECTED		
Charlotte Seltzer, Creedmoor Psychiatric Center, 80-45 Winchester Blvd., Queens Village, NY 11427, 718-264-4552	NY	08/02/94
ETA CONTROL NUMBER—1/213583 ACTION—ACCEPTED		
Kenneth M. Brown, Margaret Tietz Center/Nursing Care, 164-11 Chapin Parkway	NY	08/02/94
Stony Brook University Hospital, Health Science Center, L-3, Rm. 106, Stony Brook, NY 11794-8300, 516-444-2525	NY	08/01/94
ETA CONTROL NUMBER—1/213526 ACTION—ACCEPTED		
Robert Koenig, Woodmere Health Care & Nursing Fac, 130 Irving Place, Woodmere, NY 11598, 516-374-9300	NY	08/02/94
ETA CONTROL NUMBER—1/213598 ACTION—ACCEPTED		
ETA REGION 1 08/08/94 TO 08/14/94		
Michele B. Anderson, ARO Community Services, Inc., 11 Northeastern Blvd., Nashua, NH 03062-3139, 603-598-9800.	NH	08/09/94
ETA CONTROL NUMBER—1/213740 ACTION—ACCEPTED		
Maryann Dolak, Hudson Management Consultants, Inc., 50 Maine Avenue, Rockville Centre, NY 11570, 516-536-8000.	NY	08/09/94
ETA CONTROL NUMBER—1/213742 ACTION—ACCEPTED		
ETA REGION 1 08/15/94 TO 08/21/94		
Linda Shyavitz, Sturdy Memorial Hospital, Inc., 211 Park Street, P.O. Box 2963, Attleboro, MA 02703, 508-222-5200	MA	08/16/94
ETA CONTROL NUMBER—1/213909 ACTION—ACCEPTED		
Richard Courville, Mammoth Nursing Home, 1, Mammoth, Manchester, NH 03109, 603-625-9891	NH	08/16/94
ETA CONTROL NUMBER—1/213915 ACTION—ACCEPTED		
Edward Zirbser, Greenbriar Nursig Ctr of Hammonton, 190 N. Evergreen Avenue, Woodbury, NJ 08096, 609-848-7400.	NJ	08/15/94
ETA CONTROL NUMBER—1/213887 ACTION—ACCEPTED		
Magdy Elamir, Jersey City Neurological Center, 550 Summit Avenue, Jersey City, NJ 07306, 201-653-0022	NJ	08/16/94
ETA CONTROL NUMBER—1/213911 ACTION—ACCEPTED		
C. Beth Kelly, Lakewood Nursing Center, 285 River Avenue, Lakewood, NJ 08701, 908-363-0400	NJ	08/15/94
ETA CONTROL NUMBER—1/213886 ACTION—ACCEPTED		
Blanche Bonifacio, Merry Heart Nursing Home, 200 Route 10, Succasunna, NJ 07876, 201-584-4000	NJ	08/16/94
ETA CONTROL NUMBER—1/213912 ACTION—ACCEPTED		
Carmen B. Alecci, West Hudson Hospital, 206 Bergen Ave, Kearney, NJ 07032, 201-955-7014	NJ	08/15/94
ETA CONTROL NUMBER—1/213888 ACTION—ACCEPTED		
Frank Maddalena, Brookdale Hospital Medical Center, Linden Boulevard/Brookdale Plaza, Brooklyn, NY 11218-3198, 718-240-5058.	NY	08/16/94
ETA CONTROL NUMBER—1/213910 ACTION—ACCEPTED		
ETA REGION 1 08/22/94 TO 08/28/94		
Scott L. Goldberg, MediCenter of Lakewood, 685 River Ave., Lakewood, NJ 08701, 908-364-8300	NJ	08/23/94
ETA CONTROL NUMBER—1/214133 ACTION—ACCEPTED		
ETA REGION 10 07/11/94 TO 07/17/94		
Michael Freeman, Bullhead Community Hospital, 2735 Silver Creek Road, Bullhead, AZ 86442, Bullhead, AZ 86442, 602-763-2273.	AZ	07/15/94
ETA CONTROL NUMBER—10/204726 ACTION—ACCEPTED		
Michael Freeman, Silver Ridge Village, 2812 Silver Creek Rd., Bullhead, AZ 86442, 602-763-1404	AZ	07/15/94
ETA CONTROL NUMBER—10/204725 ACTION—ACCEPTED		
Jose S Valdomar, Los Palos Convalescent Hospital, 1430 West Sixth Street, San Pedro, CA 90732, 310-832-6431	CA	07/11/94
ETA CONTROL NUMBER—10/204805 ACTION—ACCEPTED		
Teresita Nery, MedPro Home Health, Inc., 3345 Wilshire Blvd., Suite 515, Los Angeles, CA 90010, 213-384-3800	CA	07/15/94
ETA CONTROL NUMBER—10/204804 ACTION—ACCEPTED		
S. Lynn Cook, San Joaquin General Hospital, P.O. Box 1020, Stockton, CA 95201, 209-468-6260	CA	07/15/94
ETA CONTROL NUMBER—10/204751 ACTION—ACCEPTED		
Jose S. Valdomar, Seacrest Convalescent Hospital, 1416 West Sixth Street, San Pedro, CA 90732, 310-833-3526	CA	07/15/94
ETA CONTROL NUMBER—10/204750 ACTION—ACCEPTED		
ETA REGION 10 07/18/94 TO 07/24/94		
Grant Asay, St. Ann's Nursing Home, 415 Sixth Street, Juneau, AK 99801, 907-586-3383	AK	07/18/94
ETA CONTROL NUMBER—10/204786 ACTION—ACCEPTED		
Deenah Stockton, Good Samaritan Hospital, 901 Olive Drive, Bakersfield, CA 93308, 805-399-4461	CA	07/20/94

DIVISION OF FOREIGN LABOR CERTIFICATIONS, HEALTH CARE FACILITY ATTESTATIONS—Continued
[FORM ETA-9029]

CEO-Name/Facility Name/Address	State	Action date
ETA CONTROL NUMBER—10/204835 ACTION—ACCEPTED		
Joan Barlow, Placerville Pines Conv. Hospital, 1040 Marshall Way, Placerville, CA 95667, 916–622–3400 ETA CONTROL NUMBER—10/204860 ACTION—ACCEPTED	CA	07/21/94
ETA REGION 10 07/25/94 TO 07/31/94		
Anelli Stamm, Silver Oak Manor, 788 Holmes Street, Livermore, CA 94550, 510–447–2280 ETA CONTROL NUMBER—10/204956 ACTION—ACCEPTED	CA	07/28/94
ETA REGION 10 08/08/94 TO 08/14/94		
Cecil Mays, Care West Arroyo Vista, 3022 45th Street, San Diego, CA 92105, 619–283–5855 ETA CONTROL NUMBER—10/205008 ACTION—ACCEPTED	CA	08/11/94
Cecil Mays, Care West Tri City, 3232 Thunder Drive, Oceanside, CA 92056, 619–724–2183 ETA CONTROL NUMBER—10/205009 ACTION—ACCEPTED	CA	08/11/94
Leila Knox, Casa Metro Convalescent Hospital, 2020 North Weber, Fresno, CA 93105, 209–237–0883 ETA CONTROL NUMBER—10/204994 ACTION—ACCEPTED	CA	08/11/94
Karen G. Sell, Hanford Community Medical Center, 450 Greenfield Avenue, Hanford, CA 93230, 209–585–5463 ETA CONTROL NUMBER—10/204991 ACTION—ACCEPTED	CA	08/11/94
Cecil Mays, Palomares Nursing & Rehabilitation, 250 West Artesia Street, Pomona, CA 91768, 909–623–3564 ETA CONTROL NUMBER—10/205005 ACTION—ACCEPTED	CA	08/11/94
Cecil Mays, Vista Knoll, 2000 Westwood Road, Vista, CA 92083, 619–630–2273 ETA CONTROL NUMBER—10/205007 ACTION—ACCEPTED	CA	08/11/94
ETA REGION 10 08/15/94 TO 08/21/94		
Cecil Mays, Care West Anza, 622 South Anza Street, El Cajon, CA 92020, 619–442–0544 ETA CONTROL NUMBER—10/205028 ACTION—ACCEPTED	CA	08/16/94
Cecil Mays, Care West Arizona Nursing Center, 1330 17th Street, Santa Monica, CA 90404, 310–829–5411 ETA CONTROL NUMBER—10/205025 ACTION—ACCEPTED	CA	08/16/94
Cecil Mays, Care West Bayside Nursing Center, 1251 South Eliseo Drive, Kentfield, CA 94904, 415–461–1900 ETA CONTROL NUMBER—10/205023 ACTION—ACCEPTED	CA	08/16/94
Cecil Mays, Care West Gateway Nursing Center, 26660 Patrick Avenue, Hayward, CA 94554, 510–782–1845 ETA CONTROL NUMBER—10/205012 ACTION—ACCEPTED	CA	08/16/94
Cecil Mays, Care West Intercommunity Nursing, 12527 Studebaker Road, Norwalk, CA 90650, 310–868–4767 ETA CONTROL NUMBER—10/205013 ACTION—ACCEPTED	CA	08/16/94
Cecil Mays, Care West Madison Nursing & Rehab, 1391 East Madison Avenue, El Cajon, CA 92021, 619–444–1107 ETA CONTROL NUMBER—10/205011 ACTION—ACCEPTED	CA	08/16/94
Cecil Mays, Care West Manteca Nursing & Rehab, 410 Eastwood Avenue, Manteca, CA 95336, 209–239–1222 ETA CONTROL NUMBER—10/205010 ACTION—ACCEPTED	CA	08/16/94
Johntai E. Jackson, II, Ph.D., Julia's Nursing Service Agency, Suite 510, 1800 N. Argyle Avenue, Los Angeles, CA 90028, 213–466–8930. ETA CONTROL NUMBER—10/204952 ACTION—ACCEPTED	CA	08/17/94
Cecil Mays, Playa Del Rey Rehab & Care Center, 7716 Manchester Avenue, Playa del Rey, CA 90293, 310–823–4694. ETA CONTROL NUMBER—10/205027 ACTION—ACCEPTED	CA	08/16/94
Jay Jeffers, BHC Health Services oc Navada, Inc, 1240 East Ninth Street, Reno, NV 89512, 702–323–0478 ETA CONTROL NUMBER—10/204992 ACTION—ACCEPTED	NV	08/16/94
ETA REGION 5 07/04/94 TO 07/10/94		
Gary R. House, Community Care at Colorado Springs, 110 W. Van Buren St., Colorado Springs, CO 80907–8400, 719–475–8686. ETA CONTROL NUMBER—5/226722 ACTION—ACCEPTED	CO	07/05/94
Lynn Bunkholder, Community Care at Panoia, 1625 Meadowbrook Blvd., Paonia, CO 81428, 303–527–4837 ETA CONTROL NUMBER—5/226709 ACTION—ACCEPTED	CO	07/05/94
Lucie Frah, Community Care of America, 2825 Patterson Road, Grand Junction, CO 81506, 303–342–7356 ETA CONTROL NUMBER—5/226707 ACTION—ACCEPTED	CO	07/05/94
Paul Whisler, Community Care of Mediapolis, 608 Prairie Street, Mediopolis, IA 52637, 319–394–3991 ETA CONTROL NUMBER—5/226723 ACTION—ACCEPTED	IA	07/05/94
Charles Stumpf, Margaret Manor, Inc., 1121 N. Orleans St., Chicago, IL 60604, 312–943–4300 ETA CONTROL NUMBER—5/226671 ACTION—ACCEPTED	IL	07/05/94
Roberta Caurdy, Advance Nursing Center, 2936 South John Daly, Inkster, MI 48141, 313–278–7272 ETA CONTROL NUMBER—5/226665 ACTION—ACCEPTED	MI	07/05/94
Noreen Trout, Medical Case Management of America, 812 B East Franklin St., Centerville, OH 45459, 800–538–4218	OH	07/05/94

DIVISION OF FOREIGN LABOR CERTIFICATIONS, HEALTH CARE FACILITY ATTESTATIONS—Continued
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CEO-Name/Facility Name/Address	State	Action date
ETA CONTROL NUMBER—5/226727 ACTION—ACCEPTED		
ETA REGION 5 07/11/94 TO 07/17/94		
Carolyn Manna, Community Care, 139 Park Drive, Grand Junction, CO 81501, 303–260–1152 ETA CONTROL NUMBER—5/227263 ACTION—ACCEPTED	CO	07/14/94
Kelly Everly, Community Care at Delta, 2050 South Main, Delta, CO 81416, 303–874–9775 ETA CONTROL NUMBER—5/227258 ACTION—ACCEPTED	CO	07/14/94
Janet B. Ryder, Community Care at La Villa Grande, 2501 Little Bookcliff Drive, Grand Junction, CO 81501, 303–245–1211. ETA CONTROL NUMBER—5/227256 ACTION—ACCEPTED	CO	07/14/94
Larry Levelle, Community Care of Cannon City, 515 Fairview, Canon City, CO 81212, 719–275–0665 ETA CONTROL NUMBER—5/227252 ACTION—ACCEPTED	CO	07/14/94
Sharon Shumaker, Community Care of Am. at Muscatine, 3440 Mulberry Avenue, Muscatine, IA 52761, 319–263–2194. ETA CONTROL NUMBER—5/227268 ACTION—ACCEPTED	IA	07/14/94
Steven Frank, Apple Home Healthcare, Ltd., 2777 Finley Road, Suite 10, Downers Grove, IL 60515, 708–495–6060 ETA CONTROL NUMBER—5/227397 ACTION—ACCEPTED	IL	07/15/94
Bryan Barrish, Elmwood Care, Inc., 7733 West Grand Avenue, Elmwood Park, IL 60635, 708–452–9200 ETA CONTROL NUMBER—5/227399 ACTION—ACCEPTED	IL	07/15/94
Chung S. Kim, M.D., KBC Health Centre, Inc., d/b/a Lake Bluff HealthCare Centre, 700 Jenkisson Avenue, Lake Bluff, IL 60044, 708–295–3900. ETA CONTROL NUMBER—5/227398 ACTION—ACCEPTED	IL	07/15/94
Marlaine Brunslik, Loretto Hospital, 645 South Central Avenue, Chicago, IL 60644, 312–854–5044 ETA CONTROL NUMBER—5/227261 ACTION—ACCEPTED	IL	07/14/94
Perla J. Cordero, New Life Health Care Personnel, 651 W. Gladys St., Elmhurst, IL 60126–1874, 708–530–5170 ETA CONTROL NUMBER—5/227396 ACTION—ACCEPTED	IL	07/15/94
Barbara H. Hecht, Regency Nursing Center, 6631 Milwaukee Avenue, Niles, IL, 708–647–7444 ETA CONTROL NUMBER—5/227394 ACTION—ACCEPTED	IL	07/15/94
Celia Anschutz, Christopher Manor of Lucas, 414 North Main P.O. Box 68, Lucas, KS 67648, 913–525–6215 ETA CONTROL NUMBER—5/227273 ACTION—ACCEPTED	KS	07/14/94
Shirley Robinson, Community Care of America, Inc., 117 W. First Street, P.O. Box 369, Smith Center, KS 66967, 913–282–6696. ETA CONTROL NUMBER—5/227248 ACTION—ACCEPTED	KS	07/14/94
Salvatore Bensiatto, Eastwood Nursing Center, 626 East Grand Boulevard, Detroit, MI 48207, 313–923–5816 ETA CONTROL NUMBER—5/227401 ACTION—ACCEPTED	MI	07/15/94
Salvatore Bensiatto, Father Solanus Casey Nursing Ctr., 660 East Grand Boulevard, Detroit, MI 48207, 313–923–5800 ETA CONTROL NUMBER—5/227402 ACTION—ACCEPTED	MI	07/15/94
Salvatore Benisatto, Westwood Nursing Center, 16588 Schaefer, Detroit, MI 48235, 313–345–5000 ETA CONTROL NUMBER—5/227400 ACTION—ACCEPTED	MI	07/15/94
Steve Riely, Community Care of Amer. at Tarkio, 300 Cedar, Tarkio, MO 64491, 816–736–4116 ETA CONTROL NUMBER—5/227260 ACTION—ACCEPTED	MO	07/14/94
John Turner, Community Care at Ashland, 1700 Furnas Street, Ashland, NE 68003, 402–944–7031 ETA CONTROL NUMBER—5/227271 ACTION—ACCEPTED	NE	07/15/94
Christi Karle, Community Care at Edgar, Route 1, P.O. Box 1183, Edgar, NE 68935, 402–224–5015 ETA CONTROL NUMBER—5/227247 ACTION—ACCEPTED	NE	07/14/94
Peggy Ryan, Community Care at Sutherland, 333 Maple Street, Sutherland, NE 69165, 308–386–4393 ETA CONTROL NUMBER—5/227257 ACTION—ACCEPTED	NE	07/14/94
Joyce Bauer, Community Care of Ainsworth, 143 No. Fullerton, Ainsworth, NE 69210, 402–387–2500 ETA CONTROL NUMBER—5/227254 ACTION—ACCEPTED	NE	07/14/94
Connie Jones, Community Care of Amer. at Aurora, 616 13th Street, P.O. Box 266, Aurora, NE 68818, 402–694–6905 ETA CONTROL NUMBER—5/227264 ACTION—ACCEPTED	NE	07/14/94
Marcia Malone, Community Care of Amer. at Waverly, 11041 North 137th Street, P.O. Box 160, Waverly, NE 68462, 402–786–2626. ETA CONTROL NUMBER—5/227249 ACTION—ACCEPTED	NE	07/14/94
Lyn Hemphill, Community Care of Utica, 1350 Centennial Avenue, Utica, NE 68456, 402–534–2041 ETA CONTROL NUMBER—5/227251 ACTION—ACCEPTED	NE	07/14/94
Tamara Schell, Grandview Manor, Broad Street & Highway 4, Campbell, NE 68932, 402–756–8701 ETA CONTROL NUMBER—5/227246 ACTION—ACCEPTED	NE	07/14/94
Michael Garnet, Community Care of Amer. at Waland, 1901 Howell, Waland, WY 82401, 307–347–4285 ETA CONTROL NUMBER—5/227266 ACTION—ACCEPTED	WY	07/14/94
ETA REGION 5 07/18/94 TO 07/24/94		
Jeff White, Bloomingdale Pavilion Inc., 311 Edgewater Drive, Bloomingdale, IL 60108, 708–894–7400 ETA CONTROL NUMBER—5/227447 ACTION—ACCEPTED	IL	07/18/94
Dov Solomon, Lincoln Park Terrace, Inc., 2732 N. Hampden Court, Chicago, IL 60614 312–248–6000 ETA CONTROL NUMBER—5/227588 ACTION—ACCEPTED	IL	07/20/94
Cynthia Sauer, Wellington Plaza Nursing Center, 504 W. Wellington Avenue, Chicago, IL 60657, 312–281–6200	IL	07/19/94

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CEO-Name/Facility Name/Address	State	Action date
ETA CONTROL NUMBER—5/227576 ACTION—ACCEPTED		
Bonnie Alterwitz, Johns Hopkins Hospital, Office of Career Services, Rm. 300, Houck Bldg., 600 North Wolfe St., Baltimore, MD 21287, 410–955–6529.	MD	07/19/94
ETA CONTROL NUMBER—5/227590 ACTION—ACCEPTED		
ETA REGION 5 07/25/94 TO 07/31/94		
Jakob Bakst, Hillcrest Healthcare Center, Inc., 777 Draper Avenue, Joliet, IL 60432, 815–727–4794	IL	07/26/94
ETA CONTROL NUMBER—5/227918 ACTION—ACCEPTED		
M. Mermelstein, Lake Front Healthcare Center, Inc., 7618 N. Sheridan, Chicago, IL 60626, 312–743–7711	IL	07/26/94
ETA CONTROL NUMBER—5/227939 ACTION—ACCEPTED		
Mr. Pat Owen or Jill Henson, Walnut Ridge Healthcare & Rehab, 555 West Carpenter Street, Springfield, IL 62702, 217–525–1880.	IL	07/29/94
ETA CONTROL NUMBER—5/228099 ACTION—ACCEPTED		
Marshall Mauer, Woodbridge Nursing Pavilion, Ltd., 2242 N. Kedzie Avenue, Chicago, IL 60647, 708–679–6725	IL	07/29/94
ETA CONTROL NUMBER—5/228080 ACTION—ACCEPTED		
Sharon D. McKenzie, Experts In Home Health Management, 25150 Evergreen, Southfield, MI 48075, 810–353–4663	MI	07/29/94
ETA CONTROL NUMBER—5/228074 ACTION—ACCEPTED		
David S. Midenberg, Lakeland Convalescent Center, Inc., P.O. Box 189 751 E. Grand Blvd., St. Clair Shores, MI 48080, 313–921–0998.	MI	07/29/94
ETA CONTROL NUMBER—5/228100 ACTION—ACCEPTED		
ETA REGION 5 08/01/94 TO 08/07/94		
Marian Stevenson, Comm. Care of America at Toledo, P.O. Box 279 Grandview Drive, Toledo, IA 52342, 515–484–5080.	IA	08/05/94
ETA CONTROL NUMBER—5/228435 ACTION—ACCEPTED		
Demi Rafael, Health Services Specialist, Inc., 1880 Spruce Avenue, Highland Park, IL 60035, 708–831–1356	IL	08/05/94
ETA CONTROL NUMBER—5/228439 ACTION—ACCEPTED		
Virginia Moravetz, Bethany Care Center, 42235 C.R. 390, Bloomingdale, MI 49026, 616–521–3383	MI	08/05/94
ETA CONTROL NUMBER—5/228432 ACTION—ACCEPTED		
ETA REGION 5 08/08/94 TO 08/14/94		
Diane Rucker, Chevy Chase Nursing Center, 3400 S. Indiana, Chicago, IL 60616, 312–842–5000	IL	08/12/94
ETA CONTROL NUMBER—5/228799 ACTION—ACCEPTED		
Julie Capouch, Elmwood Nursing and Rehab. Center, 1017 W. Galena Blvd., Aurora, IL 60506, 708–897–3100	IL	08/12/94
ETA CONTROL NUMBER—5/228811 ACTION—ACCEPTED		
Felice Cordero, Elston Nursing Center, 4340 North Keystone, Chicago, IL 60641, 312–545–8700	IL	08/10/94
ETA CONTROL NUMBER—5/228713 ACTION—ACCEPTED		
Felice Cordero, Glen Oaks Nursing Center, 270 Skokie Hwy., Northbrook, IL 60062, 708–498–9320	IL	08/12/94
ETA CONTROL NUMBER—5/228762 ACTION—ACCEPTED		
Felice Cordero, GlenBridge Nursing & Rehab. Ctr., 8333 West Golf Road, Niles, IL 60648, 708–966–9190	IL	08/11/94
ETA CONTROL NUMBER—5/228761 ACTION—ACCEPTED		
Felice Cordero, GlenShire Nursing & Rehab. Center, 22660 So. Cicero Avenue, Richton Park, IL 60471, 708–747–6120.	IL	08/11/94
ETA CONTROL NUMBER—5/228760 ACTION—ACCEPTED		
Jakob Bakst, Imperial of Hazel Crest, Inc., 3300 West 175th Street, Hazel Crest, IL, 708–335–2400	IL	08/12/94
ETA CONTROL NUMBER—5/228810 ACTION—ACCEPTED		
Virgie Taberos or Quinn Corcoran, Maplewood Health Care Center, 310 Banbury Road, North Aurora, IL 60542, 708–892–7627.	IL	08/11/94
ETA CONTROL NUMBER—5/228758 ACTION—ACCEPTED		
Ross Brown, Oakwood Terrace, 1300 Oak Avenue, Evanston, IL 60201, 708–869–1300	IL	08/12/94
ETA CONTROL NUMBER—5/228802 ACTION—ACCEPTED		
Lucille Devaux, Royal Terrace Healthcare Center, 803 Royal Drive, McHenry, IL 60050, 815–344–2600	IL	08/12/94
ETA CONTROL NUMBER—5/228808 ACTION—ACCEPTED		
Morris Esformes, West Chicago Terrace, 928 Joliet Street, West Chicago, IL 60185, 708–231–9292	IL	08/12/94
ETA CONTROL NUMBER—5/228809 ACTION—ACCEPTED		
Judy Reitz or Ronald Peterson, Johns Hopkins Bayview Medical Cntr, 4940 Eastern Avenue, Baltimore, MD 21224, 410–550–0126.	MD	08/10/94
ETA CONTROL NUMBER—5/228707 ACTION—ACCEPTED		
Nancy L. Furbish, Alpha Annex Nursing Home, 609 E. Grand Blvd., Detroit, MI 48207, 313–923–8262	MI	08/12/94
ETA CONTROL NUMBER—5/228813 ACTION—ACCEPTED		
Jess Boyer, HealthSpring, Inc., 11921 Freedom Drive, Ste. 600, Reston, VA 22090, 703–834–5646	VA	08/10/94

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CEO-Name/Facility Name/Address	State	Action date
ETA CONTROL NUMBER—5/228716 ACTION—ACCEPTED		
ETA REGION 5 **8/15/94 TO 08/21/94**		
Ruth Bosworth, Community Care at Clarinda, 600 Manor Drive, Clarinda, IA 51632, 712–542–5161 ETA CONTROL NUMBER—5/228829 ACTION—ACCEPTED	IA	08/15/94
Wendell P. Monyak, Bohemian Home for the Aged, 5061 North Pulaski Road, Chicago, IL 60630, 312–588–1220 ETA CONTROL NUMBER—5/228825 ACTION—ACCEPTED	IL	08/15/94
Paul Richman or Leo Feigenbaum, Concord Nursing Home, 9401 S. Ridgeland Avenue, Oak Lawn, IL 60453, 708–599–6700. ETA CONTROL NUMBER—5/229030 ACTION—ACCEPTED	IL	08/17/94
Judy Pree, Gilman Nursing Home, P.O. Box 307, Route 45 South, Gilman, IL 60938, 815–265–7208 ETA CONTROL NUMBER—5/228831 ACTION—ACCEPTED	IL	08/15/94
Jeff S. Berns, Norridge Nursing Center, Inc., 7001 W. Cullom, Norridge, IL 60634, 708–457–0700 ETA CONTROL NUMBER—5/228830 ACTION—ACCEPTED	IL	08/15/94
Morris Esformes, Cedars (The), 6400 The Cedars Court, Cedar Hill, MO 63016, 314–942–2700 ETA CONTROL NUMBER—5/228826 ACTION—ACCEPTED	MO	08/15/94
V.C. Vasisth, Mount View Nursing & Rehab Ctr, Inc., 102 Chandra Drive, Duncannon, PA 17020, 717–834–4111 ETA CONTROL NUMBER—5/229032 ACTION—ACCEPTED	PA	08/17/94
ETA REGION 6 **7/11/94 TO 07/17/94**		
Ms. Judy Hensley, Hillhaven Conv. Center, 5430 Linton Blvd., Delray Beach, FL 33484–6512, 407–495–3188 ETA CONTROL NUMBER—6/218328 ACTION—ACCEPTED	FL	07/13/94
Mr. Dyer Mitchell, Munroe Regional Medical Center, 131 S.W. 15th Street, Ocala, FL 32670, 904–351–7273 ETA CONTROL NUMBER—6/218494 ACTION—ACCEPTED	FL	07/15/94
Mr. Jesse Dunwoody, Southpoint Manor, 42 Collins Avenue, Miami Beach, FL 33139, 305–672–1771 ETA CONTROL NUMBER—6/218383 ACTION—ACCEPTED	FL	07/15/94
Mr. Robert N. Helms, Jr., Transitional Hospital of Tampa, 4801 N. Howard Ave., Tampa, FL 33603, 813–874–7575 ETA CONTROL NUMBER—6/218322 ACTION—ACCEPTED	FL	07/12/94
Mr. R. Hill, Britthaven of Louisburg, Rte, 3, Box 8, Louisburg, NC 27549, 919–496–7222 ETA CONTROL NUMBER—6/218432 ACTION—ACCEPTED	NC	07/15/94
Deborah Croft, Hillhaven Rose Manor, 4230 North Roxboro Road, Durham, NC 27704, 919–477–9805 ETA CONTROL NUMBER—6/218368 ACTION—ACCEPTED	NC	07/13/94
Ms. Jean Eastwood, Meadowbrook Manor, Box 249, Clemmons, NC 27012, 910–766–9158 ETA CONTROL NUMBER—6/218367 ACTION—ACCEPTED	NC	07/13/94
Mr. Don Gray Angell, Jr., Meadowbrook Manor, Rt. 6, Box 300, Advance, NC 27006, 919–998–0240 ETA CONTROL NUMBER—6/218327 ACTION—ACCEPTED	NC	07/12/94
Mr. Cecil A. Butler, Pemberton Place Nursing Ctr., Inc., 310 East Wardell Drive, Pembroke, NC 27372–2529, 910–521–1273. ETA CONTROL NUMBER—6/218372 ACTION—ACCEPTED	NC	07/13/94
Mr. Ruben Arceo, ACE Therapy & Rehab. Clinic, Inc., Arena Tower II 7324 Southwest, Freeway, Ste. 348, Houston, TX 77074, 713–272–7844. ETA CONTROL NUMBER—6/218320 ACTION—ACCEPTED	TX	07/12/94
Mr. J. Barry Shevchuk, Houston Northwest Medical Center, 710 FM 1960 West, Houston, TX 77090, 713–440–2288 ETA CONTROL NUMBER—6/218431 ACTION—ACCEPTED	TX	07/15/94
ETA REGION 6 **07/18/94 TO 07/24/94**		
Mr. Emil Miller, CHS, Inc. Univ. Gen. Hospital, 10200 Seminole Blvd., Seminole, FL 34648, 813–545–7355 ETA CONTROL NUMBER—6/218583 ACTION—ACCEPTED	FL	07/21/94
Mr. Alan A. Fletcher, Hillhaven Rehab. Center, 4411 North Habana Avenue, Tampa, FL 33614–7299, 813–872–2771 ETA CONTROL NUMBER—6/218703 ACTION—ACCEPTED	FL	07/21/94
Mr. William A. Sanger, JFK Medical Center, 5301 S. Congress Avenue, Atlantis, FL 33462, 407–965–7300 ETA CONTROL NUMBER—6/218496 ACTION—ACCEPTED	FL	07/21/94
Ephraim Barsam, Nursing Management Services, Inc., 300 31st Street North Suite 335, St. Petersburg, FL 33713, 813–321–2411. ETA CONTROL NUMBER—6/218763 ACTION—ACCEPTED	FL	07/21/94
Dr. William Zubkoff, South Shore Hospital & Med. Ctr., 630 Alton Road, Miami Beach, FL 33139, 305–672–2100 ETA CONTROL NUMBER—6/218626 ACTION—ACCEPTED	FL	07/21/94
P. Douglas Osborne, Central State Hospital, Milledgeville, GA 31062, 912–453–4128 ETA CONTROL NUMBER—6/218788 ACTION—ACCEPTED	GA	07/21/94
Ms. Kay Beckworth, Dogwood Rehab. Center, 7560 Butner Road, Fairburn, GA 30213–1914, 404–306–7878 ETA CONTROL NUMBER—6/218624 ACTION—ACCEPTED	GA	07/21/94
Mr. Michael Mays, Twelve Oaks Health Care, 315 Upper Riverdale Road, Riverdale, GA 30274, 404–991–1050 ETA CONTROL NUMBER—6/218499 ACTION—ACCEPTED	GA	07/21/94
Mr. Bill Renick, Holly Springs Memorial Hospital, 1430 E. Salem Ave. P.O. Drawer 6000, Holly Springs, MS 38634, 601–252–1212.	MS	07/21/94

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CEO-Name/Facility Name/Address	State	Action date
ETA CONTROL NUMBER—6/218584 ACTION—ACCEPTED Mr. G. Thomas Usher, Vicksburg Medical Center, 1111 I–20 Frontage Road, Vicksburg, MS 39180, 601–636–2611	MS	07/22/94
ETA CONTROL NUMBER—6/218347 ACTION—ACCEPTED Ms. Michelle Harris, Whispering Pines, 523 Country Club Road, Fayetteville, NC 28301, 910–488–0711	NC	07/21/94
ETA CONTROL NUMBER—6/218789 ACTION—ACCEPTED Mr. James Knoble, Eastern New Mexico Medical Center, 405 W. Country Club Road, Roswell, NM 88201, 505–624–3515.	NM	07/21/94
ETA CONTROL NUMBER—6/218397 ACTION—ACCEPTED Mr. W.R. Barger, Heritage Manor of Monteagle, 218 2nd Street P.O. Box 429, Monteagle, TN 37356, 615–924–2041	TN	07/21/94
ETA CONTROL NUMBER—6/218498 ACTION—ACCEPTED Mr. James M. Flynn, Western Mental Health Institute, 11100 Hwy. 64, W. Institute, TN 38074, 901–658–5141	TN	07/21/94
ETA CONTROL NUMBER—6/218581 ACTION—ACCEPTED Mr. Don E. Miller, Fair Park Health Care Center, 2815 Martin Luther King Jr. Blvd., Dallas, TX 75215, 214–421–2159	TX	07/21/94
ETA CONTROL NUMBER—6/218586 ACTION—ACCEPTED Mr. Don E. Miller, Ferris Nurs. Care Ctr. & Retire., 201 East 5th St., Ferris, TX 75125, 214–225–5000	TX	07/21/94
ETA CONTROL NUMBER—6/218587 ACTION—ACCEPTED Mr. Neal M. Eilliott, Mountain View Place, 1600 Murchison Road, El Paso, TX 79902, 915–544–2002	TX	07/21/94
ETA CONTROL NUMBER—6/218501 ACTION—ACCEPTED Mr. Neal M. Elliott, Parkwood Place, 300 North Bynum, Lufkin, TX 75904, 409–637–7215	TX	07/21/94
ETA CONTROL NUMBER—6/218500 ACTION—ACCEPTED Mr. Don E. Miller, Rockwall Nursing Care Center, 206 Storrs, Rockwall, TX 75087, 214–771–5000	TX	07/21/94
ETA CONTROL NUMBER—6/218585 ACTION—ACCEPTED Mr. Don E. Miller, Rowlett Nursing Care Center, 9300 Highway 66, Rowlett, TX 75088, 214–475–4700	TX	07/21/94
ETA CONTROL NUMBER—6/218588 ACTION—ACCEPTED Mr. L. Marcus Fry, Jr., Sierra Medical Center, 1625 Medical Center Drive, El Paso, TX 79902, 915–747–4000	TX	07/21/94
ETA CONTROL NUMBER—6/218502 ACTION—ACCEPTED Mr. Mark Bernard, Southwest General Hospital, 7400 Barlite Blvd., San Antonio, TX 78224, 210–921–3435	TX	07/21/94
ETA CONTROL NUMBER—6/218582 ACTION—ACCEPTED Mr. Jim Bushmiaer, Stuttgart Reg. Medical Center, P.O. Box 1905, Stuttgart, AR 72160, 501–673–3511	AR	07/27/94
ETA CONTROL NUMBER—6/218854 ACTION—ACCEPTED Mr. Rick Knight, Carrollwood Care Center, 15002 Hutchinson Road, Tampa FL 33625, 813–960–1969	FL	07/28/94
ETA CONTROL NUMBER—6/218901 ACTION—ACCEPTED Ms. Carmelita P. Galang, International Med. Staffing, Inc., P.O. Box 47974, St. Petersburg, FL 33743–7974, 813–384–5902.	FL	07/27/94
ETA CONTROL NUMBER—6/218895 ACTION—ACCEPTED Mr. Daniel J. Friedrich III, Pompano Beach Medical Center, 600 S.W. Third Street, Pompano Beach, FL 33060, 305–782–2000.	FL	07/28/94
ETA CONTROL NUMBER—6/218949 ACTION—ACCEPTED Mr. Daniel J. Friedrich III, Pompano Beach Medical Center, 600 S.W. Third Street, Pompano Beach, FL 33060, 305–782–2000.	FL	07/28/94
ETA CONTROL NUMBER—6/218948 ACTION—ACCEPTED Mr. Henry Robertts, Brian Center-Austell, 2130 Anderson Mill Road, Austell, GA 30073, 404–941–8813	GA	07/28/94
ETA CONTROL NUMBER—6/218896 ACTION—ACCEPTED Mr. Stelling Nelson, Chaplinwood Nursing Home, 325 Allen Memorial Drive, Milledgeville, GA 31032, 912–453–851	GA	07–27–94
ETA CONTROL NUMBER—6/218851 ACTION—ACCEPTED Mr. Mark Jacobs, Cherokee Nursing Home, Box 937, Calhoun, GA 30701, 706–629–1289	GA	07–27–94
ETA CONTROL NUMBER—6/218852 ACTION—ACCEPTED Ms. Paulette Adams, Starcrest of Newnan, 120 Spring Street, Newnan, GA 30263, 404–253–1475	GA	07–27–94
ETA CONTROL NUMBER—6/218894 ACTION—ACCEPTED Patricia Troxell, Autumn Care of Marshville, 311 W. Phifer Street, P.O. Box 608, Marshville, NC 28103, 704–624–6643	NC	07–27–94
ETA CONTROL NUMBER—6/218839 ACTION—ACCEPTED Sharon Stiles, Brian Center, 969 Cox Road, Gastonia, NC 28054, 704–866–8596	NC	07–28–94
ETA CONTROL NUMBER—6/218840 ACTION—ACCEPTED Ms. Maxine Sasser, Brian Center-Windsor, 1306 S. King St., Windsor, NC 27983, 919–794–5146	NC	07–28–94
ETA CONTROL NUMBER—6/218900 ACTION—ACCEPTED Mr. Richard Hess, Evergreens, Inc., 4007 W. Wendaver Ave., Greensboro, NC 27407, 910–854–7122	NC	07–27–94
ETA CONTROL NUMBER—6/218850 ACTION—ACCEPTED Ms. Donna Rein, Meadowbrook Terrace of N. Raleigh, 8200 Litchford Rd., Raleigh, NC 27615, 919–878–7772	NC	07–28–94
ETA CONTROL NUMBER—6/218898 ACTION—ACCEPTED Mr. Philip Holmes, Chandler Nursing Center, 601 West First, Chandler, OK 74834, 405–258–1131	OK	07–28–94
ETA CONTROL NUMBER—6/218837 ACTION—ACCEPTED Mr. Lawrence J. Centella, REN Corporation—USA, 1326 Dow St., Murfreesboro, TN 37209, 615–353–4200	TN	07–27–94
ETA CONTROL NUMBER—6/218853 ACTION—ACCEPTED Mr. Thomas B. Symonds, Mission Hospital, Inc., 900 South Bryan Road, Mission, TX 78572, 210–580–9000	TX	07–27–94
ETA CONTROL NUMBER—6/218848 ACTION—ACCEPTED K. Stevem Rowley, South Park Hospital & Medical Ctr., 6610 Quaker Avenue, Lubbock, TX 79413, 806–791–8000	TX	07–27–94
ETA CONTROL NUMBER—6/218838 ACTION—ACCEPTED Mr. Pete T. Duarte, Thomason Hospital, 4815 Alameda Avenue, El Paso, TX 79905, 915–521–7950	TX	07–27–94

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CEO-Name/Facility Name/Address	State	Action date
ETA CONTROL NUMBER—6/218836 ACTION—ACCEPTED		
ETA REGION 6 08/01/94 TO 08/07/94		
Mr. Mark Aanonson, Osceola Regional Hospital, 700 West Oak Street, Kissimmee, FL 34741, 407–846–2266 ETA CONTROL NUMBER—6/218986 ACTION—ACCEPTED	FL	08–02–94
Mr. Richard S. Freeman, West Boca Medical Center, 21644 State Road 7, Boca Raton, FL 33428, 407–488–8000 ETA CONTROL NUMBER—6/219041 ACTION—ACCEPTED	FL	08–02–94
Ms. Myrtle Vickers, Shady Acres, Inc., 1310 W. Gordon, Douglas, GA 31533, 912–384–7811 ETA CONTROL NUMBER—6/219143 ACTION—ACCEPTED	GA	08–05–94
Mr. Gary M. Stein, Touro Infirmary, 1401 Foucher Street, New Orleans, LA 70115, 504–897–8900 ETA CONTROL NUMBER—6/218990 ACTION—ACCEPTED	LA	08/02/94
Mr. Dan Cotten, Brian Center of Wilson, P.O. Box 3566, Wilson, NC 27895, 919–237–6300 ETA CONTROL NUMBER—6/219043 ACTION—ACCEPTED	NC	08/02/94
Mr. Steve Messer, Brian Center-Charlotte/Shamrock, 2727 Shamrock Drive, Charlotte, NC 28205, 704–563–0886 ETA CONTROL NUMBER—6/219044 ACTION—ACCEPTED	NC	08/02/94
Mr. Dave Carver, Brighton Manor, 415 Sunset Dr., Fuquay-Varina, NC 27526, 919–552–5609 ETA CONTROL NUMBER—6/219104 ACTION—ACCEPTED	NC	08/02/94
Ms. Frances Messar, Carver Living, 321 E. Carver St., Durham, NC 27704, 919–471–3558 ETA CONTROL NUMBER—6/218984 ACTION—ACCEPTED	NC	08/02/94
Mr. Mel Bourne, Evangeline of Woodfin, 25 Reynolds Mtn. Blvd., Asheville, NC 28804, 704–645–6619 ETA CONTROL NUMBER—6/219145 ACTION—ACCEPTED	NC	08/05/94
Mr. Russell Myers, Hillside Nursing of Wake Forest, 968 Wait Avenue P.O. Box 1826, Wake Forest, NC 27587, 919–556–4082. ETA CONTROL NUMBER—6/218985 ACTION—ACCEPTED	NC	08/02/94
Ms. Mary T. Lennon, Len-Care Nursing/Conv. Center, Inc., Highway 701 S. P.O. Box 2310, Elizabethtown, NC 28337, 910–862–8100. ETA CONTROL NUMBER ACTION—ACCEPTED	NC	08/02/94
Mr. Harold Hunter, Jr., Williamsburg Hospital, P.O. Drawer 568, Kingstree, SC 29556, 803–354–9661 ETA CONTROL NUMBER—6/219101 ACTION—ACCEPTED	SC	08/02/94
Mr. Stephen Adams, Pebble Creek Nursing Center, 11608 Scott Simpson, El Paso, TX 79936, 915–857–0071 ETA CONTROL NUMBER—6/219146 ACTION—ACCEPTED	TX	08/05/94
Mr. Michael S. Potter, Physicians & Surgeons Hospital, 3201 Sage St., Midland, TX 79705, 915–683–2273 ETA CONTROL NUMBER—6/219148 ACTION—ACCEPTED	TX	08/05/94
Ms. Nancy Wood, Renaissance Nursing Home—Katy, 1525 Tull Drive, Katy, TX 77449, 713–578–1600 ETA CONTROL NUMBER—6/219142 ACTION—ACCEPTED	TX	08/02/94
ETA REGION 6 08/08/94 TO 08/14/94		
Mr. Davide M. Carbone, Aventura Hospital, 20900 Biscayne Boulevard, Miami, FL 33180, 305–682–7000 ETA CONTROL NUMBER—6/219199 ACTION—ACCEPTED	FL	08/10/94
Mr. Nicholas Stavropoulos, Medi-Search International, 16140 Prestwich Dr., E. Loxahachee, FL 33470, 407–798–8704 ETA CONTROL NUMBER—6/219355 ACTION—ACCEPTED	FL	08/11/94
Mr. Frank Murphy, Morton Plant Hospital, 323 Jeffords Street, P.O. Box 210, Clearwater, FL 34616, 813–462–7000 ETA CONTROL NUMBER—6/219253 ACTION—ACCEPTED	FL	08/10/94
Mr. Scott Perlman, Titusville Nursing, 1705 Jess Parish Court, Titusville, FL 32716, 305–269–5720 ETA CONTROL NUMBER—6/219353 ACTION—ACCEPTED	FL	08/11/94
Mr. William F. Borne, Analytical Nursing Mtg. Corp., 3029 S. Sherwoodforest Blvd., Suite 300, Baton Rouge, LA 70816, 504–292–2031. ETA CONTROL NUMBER—6/219292 ACTION—ACCEPTED	LA	08/11/94
Mr. Steven L. Smith, Earl K. Long Medical Center, 5825 Airline Highway, P.O. Box 52999, Baton Rouge, LA 70805, 504–358–1000. ETA CONTROL NUMBER—6/219293 ACTION—ACCEPTED	LA	08/11/94
Mr. Mary Ann Thompson, Brian Center-Lincolnton, P.O. Box 249, Lincolnton, NC 28093, 704–735–8065 ETA CONTROL NUMBER—6/219349 ACTION—ACCEPTED	NC	08/11/94
Mr. Felton Wooten, The Evergreens, 206 Greensboro Road, High Point, NC 27260, 910–886–4121 ETA CONTROL NUMBER—6/219249 ACTION—ACCEPTED	NC	08/10/94
Mr. Jack Russell, Vespera Nursing Home, 1000 College Street, Wilkesboro, NC 28697, 910–838–4141 ETA CONTROL NUMBER—6/219346 ACTION—ACCEPTED	NC	08/11/94
Mr. Elijah D. Nacionales, Good Samaritan Health & Rehab., 500 Hickory Hollow Terrace, Antioch, TN 37013, 615–731–7130. ETA CONTROL NUMBER—6/219352 ACTION—ACCEPTED	TN	08/11/94
Ms. Casilda Webb, Casha Res. Home Health Serv. Inc., 9901 E. Valley Ranch Pkwy., Suite 1040, Irving, TX 75060, 214–556–0808. ETA CONTROL NUMBER—6/219354 ACTION—ACCEPTED	TX	08/11/94
Mr. Donald A. Anderson, Everglades Memorial Hospital, 200 S. Barfield Highway, Pahokee, FL 33476–9988, 407–924–5200. ETA CONTROL NUMBER—6/219373 ACTION—ACCEPTED	FL	08/17/94
Mr Jon C. Aaron, Oakwood Terrace, 18905 N.E. 24th Avenue, N. Miami Beach, FL 33180, 305–932–6360	FL	08/18/94

DIVISION OF FOREIGN LABOR CERTIFICATIONS, HEALTH CARE FACILITY ATTESTATIONS—Continued
[FORM ETA-9029]

CEO-Name/Facility Name/Address	State	Action date
ETA CONTROL NUMBER—6/219599 ACTION—ACCEPTED		
Mr. Ralph L. Stacey, Riverside Care Center, 899 N.W. 4th St., Miami, FL 33128, 305–326–1236	FL	08/17/94
ETA CONTROL NUMBER—6/219593 ACTION—ACCEPTED		
Mr. Albert Boulenger, SMH Homestead Hospital, 160 N.W. 13 Street, Homestead, FL 33030, 305–248–3232	FL	08/18/94
ETA CONTROL NUMBER—6/219598 ACTION—ACCEPTED		
Mr. David Banks, Stonegate Rehab. & Nursing Ctr., 2021 S.W. 1st Avenue, Ocala, FL 34474, 904–629–0063	FL	08/17/94
ETA CONTROL NUMBER—6/219595 ACTION—ACCEPTED		
Ms. Nila Willhoite, Willis-Knighton Health System, 2600 Greenwood Rd., Shreveport, LA 71118, 318–632–4692	LA	08/18/94
ETA CONTROL NUMBER—6/219597 ACTION—ACCEPTED		
Ms. Crystal Sossoman, Brian Center Health & Retirement, 520 Valley Street, Statesville, NC 28677, 704–873–0517	NC	08/17/94
ETA CONTROL NUMBER—6/219507 ACTION—ACCEPTED		
Mr. Paul Babinski, Brian Center-Charlotte, 5939 Reddman Road, Charlotte, NC, 704–563–6862	NC	08/17/94
ETA CONTROL NUMBER—6/219374 ACTION—ACCEPTED		
Mr. Floyd Steinberg, Britthaven of Chapel Hill, 1716 Legion Road, Chapel Hill, NC 27514, 919–942–2280	NC	08/17/94
ETA CONTROL NUMBER—6/219504 ACTION—ACCEPTED		
Ms. Susan Macias, Rehoboth McKinley Christian HCS, 800 A Hospital Drive, Gallup, NM 87305, 505–863–7189	NM	08/17/94
ETA CONTROL NUMBER—6/219503 ACTION—ACCEPTED		
Mr. D. W. Sims, Camp Wood Convalescent Center, P.O. Box 310, Camp Wood, TX 78833, 210–597–5250	TX	08/17/94
ETA CONTROL NUMBER—6/219376 ACTION—ACCEPTED		
Mr. Stañ Weyer, Coronado Nursing Center, 223 S. Resler Drive, El Paso, TX 79912, 915–584–9417	TX	08/17/94
ETA CONTROL NUMBER—6/219378 ACTION—ACCEPTED		
Mr. David Hodgson, Doctors Hospital of Laredo, 500 E. Mann Road, Laredo, TX 78041, 210–723–1131	TX	08/17/94
ETA CONTROL NUMBER—6/219377 ACTION—ACCEPTED		
Mr. Jerry Tanqan, McAllen Good Samaritan Center, 812 Houston Avenue, McAllen, TX 78501–0279, 210–682–6331	TX	08/17/94
ETA CONTROL NUMBER—6/219506 ACTION—ACCEPTED		

ETA REGION 6
08/22/94 TO 08/28/94

Mr. Lawrence J. Centella, REN Corporation—USA, 1160 S. Sermoran Blvd., Ste. C, Orlando, FL 32807, 407–823–9533.	FL	08/24/94
ETA CONTROL NUMBER—6/219601 ACTION—ACCEPTED		
Mr. Lawrence J. Centella, REN Corporation—USA, 4141 S. Tamiami Trail, Sarasota, FL 34231, 813–924–4025	FL	08/24/94
ETA CONTROL NUMBER—6/219600 ACTION—ACCEPTED		
Mr. Lawrence J. Centella, REN Corporation—USA, Medical Office Building 11140 W., Colonial Dr., Ste. #5, Ocoee, FL 32761, 407–877–0626.	FL	08/24/94
ETA CONTROL NUMBER—6/219604 ACTION—ACCEPTED		
Mr. Lawrence J. Centella, REN Corporation—USA, 1500 N.W., 12th Ave., Ste. 106, Miami, FL 33136, 305–324–6891	FL	08/24/94
ETA CONTROL NUMBER—6/219605 ACTION—ACCEPTED		
Mr. Lawrence J. Centella, REN Corporation—USA, 1026 S. Ridgewood Avenue, Daytona Beach, FL 32114, 904–257–3211.	FL	08/24/94
ETA CONTROL NUMBER—6/219610 ACTION—ACCEPTED		
Mr. Lawrence J. Centella, REN Corporation—USA, Lucerne Medical Plaza 100 W. Gore, St., Ste. 102, Orlando, FL 32806, 407–841–8182.	FL	02/24/94
ETA CONTROL NUMBER—6/219602 ACTION—ACCEPTED		
Mr. Lawrence J. Centella, REN Corporation—USA, 1001 NW 13th Street, Boca Raton, FL 33486, 407–362–9113	FL	08/24/94
ETA CONTROL NUMBER—6/219611 ACTION—ACCEPTED		
Mr. J. David Lawrence, Jr., B–J–C Medical Center, 70 Medical Center Drive, Commerce, GA 30529, 706–335–1000	GA	08/24/94
ETA CONTROL NUMBER—6/219783 ACTION—ACCEPTED		
Mr. Bill Lang, Community Care Center, 8422 Kurthwood Road, P.O. Box 270, Leesville, LA 71446, 318–239–6578	LA	08/24/94
ETA CONTROL NUMBER—6/219780 ACTION—ACCEPTED		
Mr. Jeff Burch, Riverlands Health Care Center, 1980 River Road P.O. Drawer CC, Lutcher, LA 70071, 504–869–5725	LA	08/24/94
ETA CONTROL NUMBER—6/219781 ACTION—ACCEPTED		
Ms. Linda Howard, Carrington Place, 600 Fullwood Lane, Matthews, NC 28105, 704–841–4920	NC	08/24/94
ETA CONTROL NUMBER—6/219782 ACTION—ACCEPTED		
Ms. Linda Roberts, Hillhaven Sunnybrook, 25 Sunnybrook Road, Raleigh, NC 27610–1894, 919–231–6150	NC	08/24/94
ETA CONTROL NUMBER—6/219614 ACTION—ACCEPTED		
Mr. James P. Seward, Hillside Hospital, Inc., 1265 E. College Street, Pulaski, TN 38478, 615–363–7531	TN	08/26/94
ETA CONTROL NUMBER—6/220922 ACTION—ACCEPTED		
Mr. J. F. Adams, Medical Plaza Hospital, 1111 Gallagher Rd., Sherman, TX 75090, 903–870–7000	TX	08/24/94
ETA CONTROL NUMBER—6/219767 ACTION—ACCEPTED		
Ms. Brenda Chung, Nightingale Services, 6220 Westpark Drive, Suite #220, Houston, TX 77057, 713–780–0695	TX	08/24/94
ETA CONTROL NUMBER—6/219617 ACTION—ACCEPTED		
Mr. Eddie Kuntz, Retama Manor, 400 S. Pete Diaz, Jr. Ave., Rio Grande, TX 78582, 210–487–2513	TX	08/24/94
ETA CONTROL NUMBER—6/219766 ACTION—ACCEPTED		
Mr. Louis Robichaux, Silver Leaves Nursing/Rehab. Ctr., 505 W. Centerville, Garland, TX 75041, 214–278–3566	TX	08/24/94
ETA CONTROL NUMBER—6/219612 ACTION—ACCEPTED		
Mr. Stan Weyer, Sunset Haven Nursing Center, 9001 N. Loop Drive, El Paso, TX 79907, 915–859–1650	TX	08/24/94

DIVISION OF FOREIGN LABOR CERTIFICATIONS, HEALTH CARE FACILITY ATTESTATIONS—Continued
[FORM ETA-9029]

CEO-Name/Facility Name/Address	State	Action date
ETA CONTROL NUMBER—6/219615 ACTION—ACCEPTED Ms. Vicki Archer, Norfolk Health Care, 1005 Hampton Road, Norfolk, VA 23507, 804-623-5602 ETA CONTROL NUMBER—6/219673 ACTION—ACCEPTED	VA	08/24/94

[FR Doc. 94–22716 Filed 9–13–94; 8:45 am]
BILLING CODE 4510–30–P

NUCLEAR REGULATORY COMMISSION

Biweekly Notice

Applications and Amendments to Facility Operating LicensesInvolving No Significant Hazards Considerations

I. Background

Pursuant to Public Law 97-415, the U.S. Nuclear Regulatory Commission (the Commission or NRC staff) is publishing this regular biweekly notice. Public Law 97-415 revised section 189 of the Atomic Energy Act of 1954, as amended (the Act), to require the Commission to publish notice of any amendments issued, or proposed to be issued, under a new provision of section 189 of the Act. This provision grants the Commission the authority to issue and make immediately effective any amendment to an operating license upon a determination by the Commission that such amendment involves no significant hazards consideration, notwithstanding the pendency before the Commission of a request for a hearing from any person.

This biweekly notice includes all notices of amendments issued, or proposed to be issued from August 22, 1994 through September 1, 1994. The last biweekly notice was published on August 31, 1994 (59 FR 45015).

Notice Of Consideration Of Issuance Of Amendments To Facility Operating Licenses, Proposed No Significant Hazards Consideration Determination, And Opportunity For A Hearing

The Commission has made a proposed determination that the following amendment requests involve no significant hazards consideration. Under the Commission's regulations in 10 CFR 50.92, this means that operation of the facility in accordance with the proposed amendment would not (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety. The basis for this proposed determination for each amendment request is shown below.

The Commission is seeking public comments on this proposed determination. Any comments received within 30 days after the date of publication of this notice will be considered in making any final determination.

Normally, the Commission will not issue the amendment until the expiration of the 30-day notice period. However, should circumstances change during the notice period such that failure to act in a timely way would result, for example, in derating or shutdown of the facility, the Commission may issue the license amendment before the expiration of the 30-day notice period, provided that its final determination is that the amendment involves no significant hazards consideration. The final determination will consider all public and State comments received before action is taken. Should the Commission take this action, it will publish in the **Federal Register** a notice of issuance and provide for opportunity for a hearing after issuance. The Commission expects that the need to take this action will occur very infrequently.

Written comments may be submitted by mail to the Rules Review and Directives Branch, Division of Freedom of Information and Publications Services, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, DC 20555, and should cite the publication date and page number of this **Federal Register** notice. Written comments may also be delivered to Room 6D22, Two White Flint North, 11545 Rockville Pike, Rockville, Maryland from 7:30 a.m. to 4:15 p.m. Federal workdays. Copies of written comments received may be examined at the NRC Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC 20555. The filing of requests for a hearing and petitions for leave to intervene is discussed below.

By October 14, 1994, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's "Rules of Practice for Domestic Licensing Proceedings" in 10 CFR Part 2. Interested persons should consult a current copy of 10 CFR 2.714 which is available at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC 20555 and at the local public document room for the particular facility involved. If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

As required by 10 CFR 2.714, a petition for leave to intervene shall set forth with particularity the interest of the petitioner in the proceeding, and how that interest may be affected by the results of the proceeding. The petition should specifically explain the reasons why intervention should be permitted with particular reference to the following factors: (1) the nature of the petitioner's right under the Act to be made a party to the proceeding; (2) the nature and extent of the petitioner's property, financial, or other interest in the proceeding; and (3) the possible effect of any order which may be entered in the proceeding on the petitioner's interest. The petition should also identify the specific aspect(s) of the subject matter of the proceeding as to which petitioner wishes to intervene. Any person who has filed a petition for leave to intervene or who has been admitted as a party may amend the petition without requesting leave of the Board up to 15 days prior to the first prehearing conference scheduled in the proceeding, but such an amended petition must satisfy the specificity requirements described above.

Not later than 15 days prior to the first prehearing conference scheduled in the proceeding, a petitioner shall file a supplement to the petition to intervene which must include a list of the contentions which are sought to be litigated in the matter. Each contention must consist of a specific statement of the issue of law or fact to be raised or controverted. In addition, the petitioner shall provide a brief explanation of the bases of the contention and a concise statement of the alleged facts or expert opinion which support the contention and on which the petitioner intends to rely in proving the contention at the hearing. The petitioner must also provide references to those specific sources and documents of which the petitioner is aware and on which the petitioner intends to rely to establish those facts or expert opinion. Petitioner must provide sufficient information to show that a genuine dispute exists with the applicant on a material issue of law or fact. Contentions shall be limited to matters within the scope of the amendment under consideration. The contention must be one which, if proven, would entitle the petitioner to relief. A petitioner who fails to file such a supplement which satisfies these requirements with respect to at least one contention will not be permitted to participate as a party.

Those permitted to intervene become parties to the proceeding, subject to any limitations in the order granting leave to intervene, and have the opportunity to participate fully in the conduct of the hearing, including the opportunity to present evidence and cross-examine witnesses.

If a hearing is requested, the Commission will make a final determination on the issue of no significant hazards consideration. The final determination will serve to decide when the hearing is held.

If the final determination is that the amendment request involves no significant hazards consideration, the Commission may issue the amendment and make it immediately effective, notwithstanding the request for a hearing. Any hearing held would take place after issuance of the amendment.

If the final determination is that the amendment request involves a significant hazards consideration, any hearing held would take place before the issuance of any amendment.

A request for a hearing or a petition for leave to intervene must be filed with the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555, Attention: Docketing and Services Branch, or may be delivered to the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington DC 20555, by the above date. Where petitions are filed during the last 10 days of the notice period, it is requested that the petitioner promptly so inform the Commission by a toll-free telephone call to Western Union at 1-(800) 248-5100 (in Missouri 1-(800) 342-6700). The Western Union operator should be given Datagram Identification Number N1023 and the following message addressed to **(Project Director)**: 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, and to the attorney for the licensee.

Nontimely filings of petitions for leave to intervene, amended petitions, supplemental petitions and/or requests for a hearing will not be entertained absent a determination by the Commission, the presiding officer or the Atomic Safety and Licensing Board that the petition and/or request should be granted based upon a balancing of 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 which is available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC 20555, and at the local public document room for the particular facility involved.

Baltimore Gas and Electric Company, Docket Nos. 50-317 and 50-318, Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2, Calvert County, Maryland

Date of amendments request: August 2, 1994

Description of amendments request: The proposed amendment would revise Technical Specifications (TSs) 3.9.1 and 3.1.2.7 and the Bases to Specification 3.1.2.7. Specifically, TS 3.9.1, "Refueling Operations, Boron Concentration," would be revised to require action to restore boron concentration to within its limits in place of the current requirement to initiate and continue boration at a rate greater than or equal to 40 gpm of 2300 ppm boric acid solution or its equivalent until the boron concentration is within its limit. TS 3.1.2.7, "Borated Water Sources - Shutdown," gives the operability requirement for borated water sources including the Refueling Water Tank (RWT), in Modes 5 and 6. The minimum boron concentration is given as 2300 ppm. While this minimum value is correct for Mode 5, a larger boron concentration may be necessary in Mode 6. The RWT is the preferred borated water source for restoring the required boron concentration as required by TS 3.9.1. Therefore, the RWT boron concentration in Mode 6 should be at least be that required by TS 3.9.1. The proposed change to TS 3.1.2.7 would clarify the boron concentration requirements. In Mode 5, 2300 ppm will continue to be required. In Mode 6, the boron concentration limit for the RWT will be the boron concentration limits given in TS 3.9.1.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. Would not involve a significant increase in the probability or consequences of an accident previously evaluated.

During refueling operations, the reactivity condition of the core is maintained consistent with the initial conditions assumed for the boron dilution event in the accident analysis (Updated Final Safety Analysis Report Section 14.3) and is sufficient to ensure the core remains subcritical during core alterations. Technical Specification 3.9.1 requires that the boron concentration be maintained to ensure a k_{eff} [is less than or equal to] 0.95. Should the boron concentration drop below the Technical Specifications limit, the Action requires boration at a specified flow rate and boron concentration until the boron concentration is restored to within its limit. Refueling boron concentrations higher than the concentration specified by the Action in [Technical] Specification 3.9.1 are allowed by the Technical Specifications and clarification of the Action for that circumstance is needed. The proposed change eliminates the specified flow rate and boron concentration in the Action and substitutes a directive to immediately initiate action to restore the boron concentration to within its limits. The accident analysis does not assume a specific boration rate, but only assumes that the operator acts to terminate the dilution.

Therefore, the consequences of the event are unchanged. In addition, the proposed change revises the boron concentration limit on the Refueling Water Tank in Mode 6 to make the boron concentration limit on the tank the same as the boron concentration limit on the reactor coolant system. This will ensure that the RWT will contain water of a sufficient boron concentration to respond to a boron dilution event.

The proposed change does not change the boron concentration or shutdown margin required by [Technical] Specification 3.9.1 and continues to meet the initial conditions of the boron dilution event. Therefore, the probability of a boron dilution event is not increased. Furthermore, the revised action ensures that the appropriate actions for a boron dilution event will be taken and that a borated water source of sufficient concentration is available to respond to that event. Therefore, the consequences of a boron dilution event are not increased.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Would not create the possibility of a new or different type of accident from any accident previously evaluated.

The proposed change does not represent a significant change in the configuration or operation of the plant. The proposed actions will results in the same operator actions as the current Technical Specifications. The minimum boron concentration of the Refueling Water Tank in Mode 6 may be increased above the current value, but the concentrations will be within the analyzed maximum concentration for that tank,

Therefore, the proposed change does not create the possibility of a new or different type of accident from any accident previously evaluated.

3. Would not involve a significant reduction in a margin of safety.

The margin of safety provided by [Technical] Specification 3.9.1 is to ensure that the core remains subcritical during a boron dilution event and during core alterations. The proposed change does not alter the required shutdown margin or significantly change the actions to be taken if that shutdown margin is lost. The proposed change ensures that all assumed borated water sources will have sufficient boron concentration to respond to boron dilution event.

Therefore, the proposed change does not involve a significant reducation in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendments request involves no significant hazards consideration.

Local Public Document Room: Calvert County Library, Prince Frederick, Maryland 20678.

Attorney for licensee: Jay E. Silbert, Esquire, Shaw, Pittman, Potts and Trowbridge, 2300 N Street, NW., Washington, DC 20037.

NRC Project Director: Michael J. Case

Baltimore Gas and Electric Company, Docket Nos. 50-317 and 50-318, Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2, Calvert County, Maryland

Date of amendments request: August 2, 1994

Description of amendments request: The proposed change would revise Technical Specifications (TSs) regarding surveillances associated with the Emergency Diesel Generators (EDGs). Specifically, TS 4.8.1.1.2.d.3.c would be revised to add high crankcase pressure to the EDG trips which are verified to be automatically bypassed on a Safety Injection Actuation Signal (SIAS). In addition, a footnote would be added stating that verification of the high crankcase pressure trip bypass will not be required on a particular EDG until the modification has been completed for that EDG.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. Would not involve a significant increase in the probability or consequences of an accident previously evaluated.

The Calvert Cliffs Emergency Diesel Generators (EDGs) are used to provide electrical power for the operation of Engineered Safety Features (ESF) and safe shutdown equipment for events involving a loss of offsite power. The EDGs are also called upon to automatically start if an accident condition (SIAS) is present. In the event of an automatic start from a SIAS, the EDGs do not assume any load until the preferred, offsite power source is actually lost. On an undervoltage condition on a vital bus, the corresponding EDGs automatically start and load.

Emergency diesel generator trips are provided to initiate engine shutdown during abnormal diesel-run conditions, thereby protecting the EDGs from any resulting damage. Under emergency conditions, EDG reliability is a key accident-mitigating factor; therefore, upon receipt of a SIAS, the EDG control logic blocks two of the normal shutdown signals so that the only signals remaining are those required to prevent rapid destruction of the diesel engine. High crankcase pressure is typically not an indication of impending rapid diesel engine failure; therefore, this trip will be added to those shutdown signals bypassed on a SIAS. The proposed Technical Specification change adds the high crankcase pressure trip as one of the EDG trips verified to be bypassed by a SIAS. A high crankcase pressure condition on one EDG will not impact either of the two unaffected EDGs, or any other equipment required to mitigate accident consequences, and satisfies the single failure criteria. The manufacturer concurs with the proposed change to bypass this trip on a SIAS. In blocking this trip on a SIAS, the ultimate effect is an increase in the reliability of the effected EDG, and therefore, no increase in the consequences of a previously evaluated accident.

Additionally, the EDGs are not initiators to any previously evaluated accident. Therefore, blocking the high crankcase pressure trip on a SIAS will not increase the probability of an accident previously evaluated.

Therefore, the proposed change does not involve a significant increase to the probability or consequences of an accident previously evaluated.

2. Would not create the possibility of a new or different type of accident from any accident previously evaluated.

The function of the EDGs is to provide power to ESF and safe shutdown equipment for events involving a loss of offsite power. The proposed change does not represent a significant change in the configuration or operation of the plant; therefore, the EDGs continue to function in an accident mitigation role. The EDGs are not accident precursors, either in the current configuration, or following the modification to block the high crankcase pressure trip.

Therefore, the proposed changes do not create the possibility of a new or different type of accident from any accident previously evaluated.

3. Would not involve a significant reduction in a margin of safety.

The margin of safety credited with the EDG function associated with this change is the reliability of the EDGs following an event involving a loss of offsite power. By blocking high crankcase pressure trips on a SIAS, this change increases the likelihood that an EDG will be able to supply power when it is needed most, during a SIAS, because the probability of an unnecessary EDG shutdown is decreased. In effect, the margin of safety associated with this function, EDG reliability, is increased.

Therefore, the proposed change does not involve a significant reduction in the margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendments request involves no significant hazards consideration.

Local Public Document Room: Calvert County Library, Prince Frederick, Maryland 20678.

Attorney for licensee: Jay E. Silbert, Esquire, Shaw, Pittman, Potts and Trowbridge, 2300 N Street, NW., Washington, DC 20037.

NRC Project Director: Michael J. Case

Baltimore Gas and Electric Company, Docket Nos. 50-317 and 50-318, Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2, Calvert County, Maryland

Date of amendments request: August 4, 1994

Description of amendments request: The proposed amendment would eliminate Technical Specifications 3/4.3.3.3, 6.9.2.b, and 6.9.2.d and Bases 3/4.3.3.3 which gives requirements for seismic monitoring instrumentation. Specifically, the requirements for operation and testing of the seismic monitoring instrumentation would be relocated to the Calvert Cliffs Nuclear Power Plant Updated Final Safety Analysis Report (UFSAR) and plant procedures.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

The proposed change has been evaluated against the standards in 10 CFR 50.92 and has been determined to not involve a significant hazards consideration, in that operation of the facility in accordance with the proposed amendments:

1. Would not involve a significant increase in the probability or consequences of an accident previously evaluated.

The seismic monitoring system is used to measure the seismic response of selected Class 1 structures, provide time-history records of seismic events, and would indicate if predetermined seismic acceleration values had been exceeded. The seismic monitoring system itself has no safety function. The system measures values which are used after the fact to assess the intensity of an earthquake.

The proposed change will relocate requirements regarding the operability and testing of the seismic monitors from the Technical Specifications to the UFSAR and plant procedures. This will allow changes to the requirements to be made without Commission approval as long as the changes meet the criteria of 10 CFR 50.59. Associated Technical Specification Special Report requirements and Bases will be deleted. Changes to the seismic monitoring system requirements which do not meet the criteria of 10 CFR 50.59 must be approved by the Commission by license amendment.

The seismic monitoring system is not an initiator and does not act to minimize the consequences of any accident previously evaluated. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Would not create the possibility of a new or different type of accident from any accident previously evaluated?

The proposed relocation of seismic monitor requirements from the Technical Specifications to the UFSAR and plant procedures does not represent a change in the configuration or operation of the plant. The seismic monitoring system will continue to be controlled under 10 CFR 50.59. Associated Technical Specification Special Report requirements and Bases will be deleted. The proposed change will not add any new hardware and will not introduce any new accident initiators. Therefore, the proposed change does not create the possibility of a new or different type of accident from any accident previously evaluated.

3. Does operation of the facility in accordance with the proposed amendment involve a significant reduction in a margin of safety?

The seismic monitoring system is used to measure the response of selected Class 1 structures to seismic events. The plant is designed to withstand the loads imposed by the maximum hypothetical accident and the design seismic disturbance without loss of functions required for reactor shutdown and emergency core cooling. As a consequence, the seismic monitoring system makes no contribution to the margin of safety, and neither do the associated special reports.

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendments request involves no significant hazards consideration.

Local Public Document Room: Calvert County Library, Prince Frederick, Maryland 20678.

Attorney for licensee: Jay E. Silbert, Esquire, Shaw, Pittman, Potts and Trowbridge, 2300 N Street, NW., Washington, DC 20037.

NRC Project Director: Michael J. Case

Duke Power Company, et al., Docket Nos. 50-413 and 50-414, Catawba Nuclear Station, Units 1 and 2, York County, South Carolina

Date of amendment request: July 18, 1994

Description of amendment request: The purpose of the proposed amendment is to separate the Technical Specification (TS) into two separate volumes, one volume explicitly for Unit 1 and one volume explicitly for Unit 2. At present, each unit has a single volume of TS which contains the specifications covering both units. In anticipation of the steam generator (SG) replacement project scheduled to begin in the fall of 1994, the licensee is requesting that the TS reflect unit specific data. Since the SG project outlines a schedule for single units, the present documentation reflecting both units in one volume will make it difficult to facilitate TS changes to a single unit. The proposed TS will modify the current situation as follows:1) The pages will now contain the same information as found before with the exception of references to different units. The Unit 1 volume will only contain parameter and setpoint values applicable to Unit 1; the Unit 2 volume will only contain information applicable to Unit 2.2) The limits established by the TS (the definitions, the limiting conditions for operation, the surveillance requirements, the Bases, etc.) will be unchanged by this amendment, with the exception of (3) below. The effect of the amendment will be that the Unit 1 TS will be found only in the volume dedicated solely to Unit 1 and likewise for Unit 2. 3) TS Sections 3.0.5 and 4.0.6 will be deleted and minor editorial changes, such as the correction of misspellings and the deletion of obsolete footnotes, will be made. TS 3.0.5 and 4.0.6 define the applicability of the current joint TS volume to each unit individually. Since each unit's TS will be located in a separate volume, no statements are necessary to indicate differences in parameters between units and TS 3.0.5 and 4.0.6 may be deleted.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

The proposed amendments would not involve a significant increase in the probability or consequences of a previously evaluated accident. The separation of the existing technical specification manual into unit-specific volumes is a strictly administrative process which will not affect the probability or consequence of any accident.

They will not create the possibility of a new or different kind of accident from any accident previously evaluated. The changes do not have any impact upon the design or operation of plant equipment; therefore, they cannot serve to initiate a new type of accident.

The proposed amendments would not involve a reduction in a margin of safety. The changes would not impact the design or operation of any plant systems or components.

Based upon the preceding analysis, Duke Power Company concludes that the proposed amendments do not involve a significant hazards consideration as defined by 10 CFR 50.92.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room: York County Library, 138 East Black Street, Rock Hill, South Carolina 29730

Attorney for licensee: Mr. Albert Carr, Duke Power Company, 422 South Church Street, Charlotte, North Carolina 28242

NRC Project Director: Herbert N. Berkow

Duke Power Company, Docket Nos. 50-369 and 50-370, McGuire Nuclear Station, Units 1 and 2, Mecklenburg County, North Carolina

Date of amendment request: July 18, 1994

Description of amendment request: The purpose of the proposed amendment is to separate the Technical

Specifications (TS) into two separate volumes, one volume explicitly for Unit 1 and one volume explicitly for Unit 2. At present, each unit has a single volume of TS which contains the specifications covering both units. In anticipation of the steam generator (SG) replacement project scheduled to begin in the fall of 1994, the licensee is requesting that the TS reflect unit specific data. Since the SG project schedules SG replacement for each unit at different times, the present common TS would make it difficult to facilitate TS changes to a single unit. The proposed amendment will modify the current TS as follows:1) The pages will now contain the same information as found before with the exception of references to different units. The Unit 1 volume will only contain parameter and setpoint values applicable to Unit 1; the Unit 2 volume will only contain information applicable to Unit 2.2) The limits established by the TS (the definitions, the limiting conditions for operation, the surveillance requirements, the Bases, etc.) will be unchanged by this amendment, with the exception of (3) below. The effect of the amendment will be that the Unit 1 TS will be found only in the volume dedicated solely to Unit 1 and likewise for Unit 2.3) TS Sections 3.0.5 and 4.0.6 will be deleted and minor editorial changes, such as the correction of misspellings and the deletion of obsolete footnotes, will be made. TS 3.0.5 and 4.0.6 define the applicability of the current joint TS volume to each unit individually. Since each unit's TS will be located in a separate volume, no statements are necessary to indicate differences in parameters between units and TS 3.0.5 and 4.0.6 may be deleted.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

The proposed amendments would not involve a significant increase in the probability or consequences of a previously evaluated accident. The separation of the existing technical specification manual into unit-specific volumes is a strictly administrative process which will not affect the probability or consequence of any accident.

They will not create the possibility of a new or different kind of accident from any accident previously evaluated. The changes do not have any impact upon the design or operation of plant equipment; therefore, they cannot serve to initiate a new type of accident.

The proposed amendments would not involve a reduction in a margin of safety. The changes would not impact the design or operation of any plant systems or components.

Based upon the preceding analysis, Duke Power Company concludes that the proposed amendments do not involve a significant hazards consideration as defined by 10 CFR 50.92.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50:92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room: Atkins Library, University of North Carolina, Charlotte (UNCC Station), North Carolina 28223

Entergy Operations, Inc., et al., Docket No. 50-416, Grand Gulf Nuclear Station, Unit 1, Claiborne County, Mississippi

Date of amendment request: June 17, 1994, as supplemented by letter dated August 17, 1994.

Description of amendment request: The amendment requests the removal of license conditions for Transamerica Delaval (TDI) Emergency Diesel Generators (EDGs) associated with NUREG-1216.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. Involve a significant increase in the probability or the consequences of an accident previously evaluated:

The proposed amendment would not involve a significant increase in the probability or consequences of an accident previously evaluated. Elimination of the required teardowns and inspections has no effect on the probability of an accident occurring, because the diesel generators are not accident initiating equipment. Also, deleting the teardowns and inspections would decrease the consequences of an accident because the availability of the engines would increase as a result of the less frequent teardowns. Additionally, the high average reliability of the TDI engines would not be negatively affected due to this change. NRC research has shown there is a period of decreased reliability immediately following intrusive teardowns, (break in period), followed by a long period of high reliability.

2. Create the possibility of a new or different kind of accident from any previously evaluated:

The proposed amendment would not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed amendment will not cause any physical change to the plant or the design or operation of the diesel units.

3. Involve a significant decrease in the margin of safety.

The proposed amendment would not involve a significant reduction in a margin of safety. The proposed amendment will increase the reliability and availability of the EDGs and therefore will not result in a decrease in a margin of safety at Grand Gulf.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room: Judge George W. Armstrong Library, Post Office Box 1406, S. Commerce at Washington, Natchez, Mississippi 39120

Attorney for licensee: Nicholas S. Reynolds, Esquire, Winston and Strawn, 1400 L Street, N.W., 12th Floor, Washington, DC 20005-3502

NRC Project Director: William D. Beckner

Entergy Operations Inc., Docket No. 50-382, Waterford Steam Electric Station, Unit 3, St. Charles Parish, Louisiana

Date of amendment request: August 9, 1994

Description of amendment request: The proposed amendment would revise the technical specifications (TSs) by relocating the functions under review and audit to the Waterford 3 quality assurance program manual. The proposed change also incorporates the TS line-item-improvement of Generic Letter 93-07, "Modification Of The Technical Specification Administrative Control Requirements For Emergency And Security Plans," dated December 28, 1993. The changes are proposed to reduce regulatory burden by relocating TS requirements that are duplicated by other regulatory requirements.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

The proposed change will have no affect on design bases accidents nor will the change directly affect any material condition of the plant that could directly contribute to causing or mitigating the effects of an accident. Relocating Review and Audit functions from the TS is consistent with the NRC Final Policy Statement on Technical Specifications Improvements and will have no negative impact on plant operation or safety. Therefore, the proposed change will not involve a significant increase in the probability or consequences of any accident previously evaluated.

The proposed change will not alter the operation of the plant or the manner in which the plant is operated. The change will not involve a design change or introduce any new failure modes. Therefore, the proposed change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed change is administrative in nature. The Waterford 3 safety margins are defined and maintained by the Technical Specifications in Sections 2-5 which are unaffected. Therefore, the proposed change will not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room: University of New Orleans Library, Louisiana Collection, Lakefront, New Orleans, Louisiana 70122

Attorney for licensee: N.S. Reynolds, Esq., Winston & Strawn 1400 L Street N.W., Washington, D.C. 20005-3502

NRC Project Director: William D. Beckner

Georgia Power Company, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, City of Dalton, Georgia, Docket Nos. 50-424 and 50-425, Vogtle Electric Generating Plant, Units 1 and 2, Burke County, Georgia

Date of amendment request: August 16, 1994

Description of amendment request: The proposed changes revise VEGP Technical Specification 3/4.7.1.1 and its bases regarding the setpoint tolerance for the Main Steam Safety Valves (MSSVs).

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. The proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated. The setpoint tolerance change for the MSSVs from plus or minus 1% to +2%, -3% is intended to accommodate setpoint drift that may occur with these valves during plant operation. However, this change will not adversely affect the pressure boundary integrity or safety function of the valves. The increase in MSSV setpoint tolerance was also reviewed with respect to the accident analyses presented in the VEGP Final Safety Analysis Report (FSAR). The evaluation demonstrated that the acceptance criteria of the accident analyses continued to be met. Additionally, the radiological consequences associated with the accident analysis are unaffected by the proposed changes. Accordingly, since the performance and capability of the MSSVs will be maintained as a result of the proposed changes with no increase in radiological consequences, there will be no significant increase in the probability or consequences of an accident previously evaluated.

2. The proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed changes do not involve any change to the configuration or method of operation of any plant equipment, and no new failure modes have been defined for any plant system or component. The design basis requirement for the MSSVs will continue to be met and the structural integrity of the valves will not be challenged. Also, the setpoint tolerance change will not adversely affect the capability of the MSSVs to perform their pressure relief function to ensure the secondary side steam design pressure is not exceeded. Additionally, the as-left lift setpoints following testing of the MSSVs will continue to be within plus or minus 1% of their lift settings, further ensuring their safety function capability. Therefore, since the function of the MSSVs is unaffected by the proposed changes, the possibility of a new or different kind of accident from any accident previously evaluated is not created.

3. The proposed changes do not involve a significant reduction in a margin of safety. All applicable acceptance criteria associated with increasing the MSSV setpoint tolerance will continue to be met. This includes the structural integrity of the valves and the effect of the setpoint change on the accident analyses presented in the VEGP FSAR. Therefore, since the MSSVs remain in compliance with the appropriate codes and standards and all applicable acceptance criteria continue to be met, there will not be a significant reduction in a margin of safety.

Based on the preceding analysis, Georgia Power Company has determined that the proposed changes to the VEGP Technical Specifications will not significantly increase the probability or consequences of an accident previously evaluated, create the possibility of a new or different kind of accident than any previously evaluated, or involve a significant reduction in a margin of safety. Therefore, the proposed changes meet the requirements of 10 CFR 50.92(c) and do not involve a significant hazards consideration.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards

Local Public Document Room: Burke County Public Library, 412 Fourth Street, Waynesboro, Georgia 30830.

Attorney for licensee: Mr. Arthur H. Domby, Troutman Sanders, NationsBank Plaza, Suite 5200, 600 Peachtree Street, NE., Atlanta, Georgia 30308

NRC Project Director: Herbert N. Berkow

GPU Nuclear Corporation, et al., Docket No. 50-219, Oyster Creek Nuclear Generating Station, Ocean County, New Jersey

Date of amendment request: August 19, 1994

Description of amendment request: The amendment updates and clarifies the surveillance requirements for control rod exercising and standby liquid control pump operability testing including the bases to be consistent with Generic Letter 93-05.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. Although the surveillance requirements are lessened by these proposed changes, the changes are consistent with those found acceptable by the NRC in GL 93-05. The proposed changes have been determined to be compatible with our plant operating experience. Based on these considerations, it is concluded that the changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The proposed changes do not involve physical changes to the plant or changes in plant operating configuration. The changes only involve frequency of testing required to be performed. The changes are consistent with those found acceptable by the NRC in GL 93-05. Thus, it is concluded that the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Although the surveillance requirements are lessened by these proposed changes, the changes are consistent with those found acceptable by the NRC in GL 93-05. The proposed changes have been determined to be compatible with our plant operating experience. Based on these considerations, it is concluded that the changes do not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room: Ocean County Library, Reference Department, 101 Washington Street, Toms River, NJ 08753

Attorney for licensee: Ernest L. Blake, Jr., Esquire. Shaw, Pittman, Potts & Trowbridge, 2300 N Street, NW., Washington, DC 20037.

NRC Project Director: John F. Stolz

IES Utilities Inc., Docket No. 50-331, Duane Arnold Energy Center, Linn County, Iowa

Date of amendment request: August 15, 1994

Description of amendment request: The proposed amendment would increase the allowable main steam isolation valve (MSIV) leakage and delete the Technical Specifications requirements applicable to the MSIV leakage control system.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

Description of Amendment Request: Proposed Change 1

This proposed change increases the allowable leak rate specified in Technical Specification (TS) 4.7.A.2.c.3 from 11.5 standard cubic feet per hour (scfh) for any one main steam isolation valve (MSIV) when tested at 24 psig to 100 scfh for any one MSIV with a total maximum pathway leakage rate of 200 scfh through all four main steam lines when tested at 24 psig. If an MSIV exceeds 100 scfh, it will be restored to less than or equal to 11.5 scfh.

Basis for proposed no significant hazards consideration determination:

1. The change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The proposed amendment does not involve a change to structures, components, or systems which would affect the probability of an accident previously evaluated in the DAEC Updated Final Safety Analysis Report (UFSAR). It results in acceptable radiological consequences for the design basis loss of coolant accident (LOCA) which was previously evaluated in the UFSAR.

Plant specific radiological analyses have been performed to assess the effects of the proposed increase in the allowable MSIV leak rate in terms of control room, technical support center (TSC), and offsite doses following a postulated design basis LOCA. These analyses utilize the hold-up volumes of the main steam piping and condenser as an alternate method for treating MSIV leakage. The radiological analyses use standard conservative assumptions for the release of source terms consistent with Regulatory Guide 1.3, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors," Revision 2, dated June 1974.

Dose contributions from the proposed MSIV leakage rate limit of 100 scfh per MSIV (with a maximum pathway leakage rate not to exceed 200 scfh through all four main steam lines) were calculated. The analysis demonstrated that the dose contributions from the proposed MSIV leakage rate resulted in an acceptable increase to the LOCA doses previously evaluated against the regulatory limits for the offsite, control room, and TSC doses as contained in 10 CFR 100 and 10 CFR 50, Appendix A (General Design Criterion 19). The revised LOCA doses are the LOCA doses previously evaluated in the UFSAR plus the MSIV leakage doses calculated assuming use of the alternate treatment method. Table 1 of Attachment 2 shows the previously calculated doses and the newly calculated doses.

It is important to note that the resulting doses are dominated by the organic iodine fractions which occur because of the conservative source term assumptions used in this analysis. For a total leakage rate of 200 scfh through all four main steam lines, more than 90 percent of the offsite, control room, and TSC iodine doses are due to the organic iodine from the Regulatory Guide 1.3 source term and organic iodine converted from the elemental iodine deposited in main steam piping systems. If the actual iodine composition from the fuel release (cesium iodine) is used in the calculations, essentially all of this organic iodine dose would be eliminated.

The TSC doses due to MSIV leakage are especially conservative. It is not expected that there will be any radioactive releases to the TSC due to MSIV leakage during the initial stages of a LOCA since it would take considerable time for the MSIV leakage to travel through the main steam lines and main steam line drain system to the condenser, into the turbine building, and finally to the atmosphere and TSC. It was conservatively estimated that the 30-day integrated dose to personnel in the TSC would increase by only 0.02 rem. The dose calculations were performed using control room occupancy factors specified in NUREG-0800, Standard Review Plan (SRP) Section 6.4.

Therefore, we conclude that the proposed change will not significantly increase the probability or consequences of any previously analyzed accidents.

2. The proposed change will not create the possibility of a new or different kind of accident from any previously evaluated. The BWROG evaluated MSIV leakage performance and concluded that MSIV leakage rates up to 100 scfh will not inhibit the capability and isolation performance of the valves to isolate the primary containment. There is no new modification to the MSIVs which could impact their operability. The LOCA has been analyzed using the main steam piping and condenser as a treatment method to process MSIV leakage at the proposed maximum rate of 200 scfh through all four main steam lines. Therefore, the proposed change will not create any new or different kind of accident from any accident previously analyzed in the UFSAR.

3. Operation of the DAEC in accordance with the proposed change will not involve a significant reduction in the margin of safety. The allowable leak rate limit specified for the MSIVs is used to quantify a maximum amount of bypass leakage assumed in the LOCA radiological analysis. Results of the analysis are evaluated against the dose requirements contained in 10 CFR 100 for the offsite doses and 10 CFR 50, Appendix A (General Design Criterion 19) for the control room and TSC doses.

The margins of safety are not significantly affected because the dose levels remain well below the limits of 10 CFR 100 and General Design Criterion 19. Therefore, the proposed change does not involve a significant reduction in the margin of safety at the DAEC.

Description of Amendment Request: Proposed Change 2

This proposed change to delete TS 3.7.E and 4.7.E and Bases section 3.7.E and 4.7.E involves eliminating the MSIV leakage control system (LCS) requirements from the TS.

1. The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. As currently described in the UFSAR, the LCS is manually initiated after a design basis LOCA occurs. Since the LCS is operated only after an accident has occurred, this proposed amendment has no effect on the probability of an accident. The proposed change results in acceptable radiological consequences of the design basis LOCA previously evaluated in the UFSAR.

The DAEC has an inherent MSIV leakage treatment capability. IES Utilities Inc. proposes to use the main steam line drains and condenser as an alternative to the LCS. Figure 1.1 of Attachment 2 shows the primary and alternate drain paths. The proposed primary drain path at DAEC employs an MSL drain downstream of the MSIVs. There are two motor-operated valves (MOVs) in series in this line between the MSL and the main condenser. Both valves must be open to establish the required drain path. Both MOVs will be provided with essential power to assure that they can be opened following the DBA LOCA to establish a large enough drain path to support the radiological analysis.

An alternate drain path will be available to convey MSIV leakage to the isolated condenser if either MOV fails to open. The alternate drain path consists of the bypass lines around the MOVs in the primary drain path. This alternate path contains a "fail open" valve and a restricting orifice. Consequently, if either primary MOV failed to open as required, the second drain path would be available to convey MSIV leakage to the main condenser. Radiological dose calculations have been performed for this alternate path as well as for the primary path. The results were acceptable. IES Utilities Inc. will update DAEC procedures as necessary to address the applicable alternate leakage treatment methods.

IES Utilities Inc. contracted with EQE Engineering Consultants (EQE) to confirm the seismic capability of the DAEC's main steam piping and condenser to serve as an alternate leakage treatment system. Seismic verification walkdowns were performed to assure that the MSLs, the steam drain lines, the condenser, and interconnecting piping and equipment that were not seismically analyzed fall within the bounds of the design characteristics of the seismic experience database as discussed in Section 6.7 of the BWROG report.

The DAEC main steam lines, main steam drain lines, condenser, and applicable interconnecting piping and equipment, are well represented by the earthquake experience data demonstrating good seismic performance, are confirmed to exhibit excellent resistance to damage from a design basis earthquake and have been shown to have substantial margin for seismic capability. The outliers that were identified are discussed in Attachment 7. They have been either evaluated to demonstrate their acceptability as they currently exist, or plant modifications will be implemented to resolve the concerns. By taking the measures discussed in Attachment 7 to ensure resolution for all of the identified outliers, IES Utilities Inc. is assured that the damage reported for the database components should not occur to the DAEC main steam piping and condenser or to the associated support systems.

Therefore, the proposed method for MSIV leakage treatment is seismically adequate to withstand the DAEC design basis earthquake and maintain pressure retaining integrity and serve as an acceptable alternative to the currently installed LCS. The capability of the alternate MSIV leakage treatment system to withstand the effects of the safe shutdown earthquake and continue to perform its intended function (treatment of MSIV leakage) satisfies the intent of the seismic requirement of Appendix A to 10 CFR 100.

Plant specific radiological analyses have been performed to assess the effects of MSIV leakage in terms of control room and offsite doses following a postulated design basis LOCA. While not previously considered a requirement for the design of the LCS, dose calculations were also performed for the TSC. These analyses utilize the hold-up volumes of the main steam piping and condenser as an alternate treatment method for the MSIV leakage. The analysis demonstrates that the proposed change results in an acceptable increase in the radiological consequences of a LOCA previously evaluated in the UFSAR. The LOCA previously evaluated in the UFSAR is still the bounding accident; the proposed change will not involve a significant increase in the consequences of an accident previously analyzed.

The LCS lines will be disconnected, capped and welded, ensuring that the integrity of the primary containment is maintained. IES Utilities Inc. will incorporate the alternate leakage treatment system into the inservice inspection (ISI) and inservice testing (IST) programs, consistent with program requirements.

2. The proposed change does not create the possibility of a new or different kind of accident from any previously evaluated. The purpose of the LCS is to reduce the untreated MSIV leakage when isolation of the primary coolant system and containment are required. Radiological dose contributions due to MSIV leakage are bounded by a LOCA. The LOCA has been analyzed using the main steam piping and condenser as a treatment method to process MSIV leakage at the proposed maximum rate of 100 scfh per MSIV and 200 scfh total maximum pathway leakage, and determined to be within the regulatory requirements. The LCS lines connected to the main steam lines will be permanently closed to assure the primary containment integrity, isolation, and leak testing capability are not compromised.

3. The proposed change to delete TS 3.7.E and 4.7.E and Bases section 3.7.E and 4.7.E does not involve a significant reduction in the margin of safety. The intended function of the LCS for treatment of MSIV leakage will be performed by using the more effective alternate path via the main steam drain lines and condenser. This treatment method is effective for treatment of MSIV leakage over an expanded leakage range. Except for the requirement to assure that certain valves are opened to establish a proper flow path from the MSIVs to the condenser and that certain valves are closed to establish the seismic boundary, the proposed method is passive and does not require any logic controls or interlocks. On the other hand, the LCS consists of complicated logic controls and sensitive equipment which must be maintained at significant cost and radiation exposure. The radiological effects on the margin of safety are discussed above for Change 1. The safety significance of the LCS in terms of public risk was addressed in NUREG/CR-4330 which contains the evaluation for eliminating the LCS and disabling the systems currently installed at BWRs. The conclusion was that the increased public risk is less than 1 percent. Therefore, the proposed change does not involve a significant reduction in the margin of safety at the DAEC.

The various attachments referred to in the above analysis may be found in the licensees request for amendment dated August 15, 1994. This document is available in the NRC's Public Document Room located at the Gelman Building, 2120 L. Street, NW., Washington, DC 20555 and at the local public document room address below.

The NRC staff has reviewed the licensee's analysis and, based on thisreview, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room: Cedar Rapids Public Library, 500 First Street, S.E., Cedar Rapids, Iowa 52401

Attorney for licensee: Jack Newman, Esquire, Kathleen H. Shea, Esquire, Newman and Holtzinger, 1615 L Street, NW., Washington, DC 20036

NRC Project Director: John N. Hannon

Northeast Nuclear Energy Company, et al., Docket No. 50-423, Millstone Nuclear Power Station, Unit No. 3, New London County, Connecticut

Date of amendment request: June 2, 1994, as supplemented August 25, 1994

Description of amendment request: The proposed amendment would change the Technical Specifications (TS) to remove expired one-time extensions of surveillances, remove an obsolete definition of charging pump operability, and incorporate 11 line item improvements in accordance with the guidance provided in Generic Letter (GL) 93-05. Other editorial changes would be made to renumber some pages and delete the blank pages from the TS.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration. The NRC staff has reviewed the licensee's analysis against the standards of 10 CFR 50.92(c). The NRC staff's review is presented below.

A. The proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated (10 CFR 50.92(c)(1)). The expired one-time extensions were in effect to September 30, 1993. Since these extensions have expired and the appropriate surveillances were performed, the proposed changes do not effect the configuration, operation, or performance of any system, or component.

The proposals to delete Definition 1.45, "THE CHARGING PUMP OPERABILITY," and modify the Index to reflect this change are administrative changes. Definition 1.45 was applicable only for cycle 4 operation. Northeast Nuclear Energy Company (NNECO) has completed the necessary modifications and no longer rely on a temporary heating source. Therefore, the elimination of Definition 1.45 does not involve a significant increase in the probability or consequences of an accident previously analyzed.

The proposed changes to incorporate the recommendations of GL 93-05 do not affect the configuration, operation or performance of the subject systems. Increasing the surveillance test intervals as proposed will reduce the number of surveillance tests and minimize the potential for inadvertent actuation of an engineered safety feature. The increase in the surveillance test intervals will enhance the operational effectiveness of plant personnel, by reducing the amount of time that the plant staff has available to perform other tasks, such as additional preventive maintenance. Additionally, increasing the surveillance test interval will reduce unnecessary wear to equipment. NNECO's proposals to delete pages that were intentionally left blank, to renumber remaining pages and renumber Sections, and modify the Index to reflect these changes are purely administrative and editorial changes. Proposals to correct typographical errors on TS pages are also administrative changes. These changes would not affect the configuration, operation, or performance of any system, structure, or component.

The proposed changes do not affect the manner by which the facility is operated and do not change any facility design feature or equipment. The proposed changes involve administrative or programmatic requirements or merely involve editorial changes, corrections, or clarifications. Since there is no change to the facility or operating procedures, there is no affect upon the probability or consequences of any accident previously analyzed.

B. The changes do not create the possibility of a new or different kind of accident from any accident previously evaluated (10 CFR 50.92(c)(2)) because they do not affect the manner by which the facility is operated and do not change any facility design feature or equipment which affects the operational characteristics of the facility. The proposed changes involve administrative or programmatic requirements or merely involve editorial changes, corrections, or clarifications.

C. The changes do not involve a significant reduction in a margin of safety (10 CFR 50.92(c)(3)) because the proposed changes do not affect the manner by which the facility is operated or involve equipment or features which affect the operational characteristics of the facility.

Based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room: Learning Resource Center, Three Rivers Community-Technical College, Thames Valley Campus, 574 New London Turnpike, Norwich, CT 06360.

Attorney for licensee: Ms. L. M. Cuoco, Senior Nuclear Counsel, Northeast Utilities Service Company, Post Office Box 270, Hartford, CT 06141-0270.

NRC Project Director: John F. Stolz

Northeast Nuclear Energy Company, et al., Docket No. 50-423, Millstone Nuclear Power Station, Unit No. 3, New London County, Connecticut

Date of amendment request: July 22, 1994

Description of amendment request: The proposed amendment would revise the Technical Specifications to incorporate a different setpoint and transient methodology for determining the maximum allowable power range neutron flux setpoint. The changes would allow Millstone Unit 3 to operate with a reduced number of main steam-line safety valves at a reduced power level, as determined by the high neutron flux setpoint.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

...The proposed changes do not involve an SHC [significant hazards consideration] because the changes would not:

1. Involve a significant increase in the probability or consequences of an accident previously evaluated.

Technical Specification Tables 3.7-1 and 3.7-2 are being revised to reflect a reduction in the maximum allowable power range neutron flux high setpoint with inoperable steam generator safety valves. The new setpoints reflect a change in the methodology for calculating the setpoints.

Westinghouse has determined that under certain conditions with typical safety analysis assumptions, the current setpoints in Tables 3.7-1 and 3.7-2 may not provide adequate steam generator overpressure protection for a Loss of Load/Turbine Trip transient at reduced power levels. At reduced power levels, a reactor trip may not be actuated early in the transient. An overtemperature delta T trip may not be generated since the core thermal margins are increased at lower power levels. The PORVs [power-operated relief valves] and pressurizer spray may control RCS [Reactor Coolant System] pressure such that a high pressurizer pressure trip isn't generated. The reactor would eventually trip on low steam generator water level, but this may not occur before steam pressure exceeds 110% of the design value if one or more MSSVs [main steam-line safety valves] are inoperable.

To address this issue, Westinghouse has developed a new method for determination of the required power range neutron flux high setpoint. The new setpoint is based upon the heat removal capability of the operable MSSVs, rather than the previous method based only on flow capacity. The new equation is shown in the proposed changes to the Technical Specification basis. This new method has been developed by Westinghouse generically and a Millstone Unit No. 3 specific calculation has been performed. The new setpoints are being incorporated in this proposed Technical Specification change.

The new method includes several conservative assumptions. The equation is developed assuming that the maximum number of inoperable MSSVs applies to each loop. For example, for four loop operation, the maximum allowable power range neutron flux high setpoint of 65% is based upon four inoperable MSSVs, one per steam generator. Thus, in the event that only one MSSV is inoperable, the application of the new setpoint is very conservative. In addition, the setpoint is based upon the assumption that the largest capacity MSSV is inoperable. For the case where one of the lower capacity MSSVs is inoperable, the setpoint will be conservative.

The method of calculating the setpoint provides assurance that the heat removal capability of the operable MSSVs is sufficient for reactor power up to the power range neutron flux high setpoint taking into account instrument and channel uncertainties. Consequently, steam generator pressure will remain below 110% of design in the event of the limiting overpressurization transient, the Loss of Load/Turbine Trip.

Reducing the power range neutron flux high setpoint and consequently the allowable reduced power level has no impact on the consequences of any other accident. In addition, since the proposed changes only involve a reduction in the allowable power range neutron flux high setpoint, and operation at a lower power level, they cannot affect the probability of any design basis accident.

2. Create the possibility of a new or different kind of accident from any previously analyzed.

Since the proposed changes just reduce the existing limit on the power range neutron flux high setpoint with inoperable MSSVs, the change cannot create the possibility for a new or different kind of accident.

3. Involve a significant reduction in the margin of safety.

The reduced setpoint provides additional assurance that the steam generator pressure will remain below 110% of design for the limiting overpressurization transient, the Loss of Load/Turbine Trip. Thus, the proposed changes do not reduce the margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room: Learning Resource Center, Three Rivers Community-Technical College, Thames Valley Campus, 574 New London Turnpike, Norwich, Connecticut 06360.

Attorney for licensee: Ms. L. M. Cuoco, Senior Nuclear Counsel, Northeast Utilities Service Company, Post Office Box 270, Hartford, Connecticut, 06141-0270.

NRC Project Director: John F. Stolz

Pennsylvania Power and Light Company, Docket No. 50-387 Susquehanna Steam Electric Station, Unit 1, Luzerne County, Pennsylvania

Date of amendment request: July 27, 1994

Description of amendment request: By letter dated June 15, 1992, Pennsylvania Power and Light Company (PP&L) submitted "Licensing Topical Report NE-092-001, Revision 0, Power Uprate With Increased Core Flow," for Susquehanna Steam Electric Station, Units 1 and 2. The report was submitted to support future amendments to the Units 1 and 2 licenses to permit a 4.5-

percent increase in reactor thermal power and an 8-percent increase in core flow for each unit. The initial submittal was revised and supplemented by letters of July 24, September 17, and December 18, 1992, and January 8, January 25, April 2, August 5, August 12, and September 29, 1993. The Commission's safety evaluation on these submittals was issued November 30, 1993 (Letter, Thomas E. Murley, NRC, to Robert G. Byram, PP&L). The Commission concluded that the revised (Revision 2) licensing topical report adequately supports PP&L's proposed power uprate. The Commission also concluded that SES, Units 1 and 2, can operate safety with the proposed 8-percent increase in core flow, the proposed 4.5-percent increase in reactor thermal power, the corresponding 5-percent increase in main turbine inlet steam flow, and the corresponding increases in flows, temperatures, pressures, and capacities required in supporting systems and components at these uprated conditions.This amendment will change several Technical Specifications sections (listed below in the no significant hazards consideration) for Susquehanna Steam Electric Station, Unit 1, to increase the licensed power level from the current 3293 MWt to a new limit of 3441 Mwt.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

The following three questions are addressed for each of the proposed Technical Specification Changes:

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

2. Does the proposed change create the possibility of a new or different kind of accident from any previously evaluated?

3. Does the proposed change involve a significant reduction in a margin of safety?

Section 1.0, Definitions, Definition 1.33, Rated Thermal Power

This change redefines Rated Thermal Power as 3441 megawatts thermal.

1. No. Neither the probability (frequency of occurrence) nor consequences of any accident previously evaluated is significantly affected by the increased power level because the design and regulatory criteria established for plant equipment remain imposed for the uprated power level. The PP&L assessment to increase the rated thermal power level at Susquehanna SES Unit 1, followed the guidelines of NEDC-31879P (≥Generic Guidelines for General Electric Boiling Water Reactor Power Uprate," G.E. Nuclear Energy, June 1991). NEDC-31879P provides generic licensing criteria, methodology, and a defined scope of analytical and equipment review to be performed to demonstrate the ability to operate safely at the uprated power level which have been approved by the NRC. NE-092-001 (≥Licensing Topical Report for Power Uprate With Increased Core Flow," Pennsylvania Power & Light Company, December 1992) provides the description of the power uprate licensing analysis methodology and the results of the evaluations performed to support the proposed uprated power operation consistent with the methodology presented in NEDC-31879P. NE-092-001 provides a description of the power uprate licensing analysis methodology which will be used to determine cycle specific thermal limits for Unit 1, Cycle 9 and future cycles and concludes that an uprated power level of 3441 megawatts thermal can be achieved without significant effect on equipment or safety analyses.

2. No. The methodology and results described above do not indicate that a possibility for a new or different kind of accident from any previously evaluated has been created by uprated operation.

3. No. Based on the response to Question 1 above, the methodology and results do not indicate a significant reduction in a margin of safety.

Section 2.1, Safety Limits

The reference to "rated core flow" in Technical Specification 2.1.1 and 2.1.2 has been replaced with a reference to actual core flow. The references to "rated core flow" have been deleted to avoid confusion since allowable core flow is being increased by 8%. 10 Mlbm/hr is being used in these specifications to be consistent with other similar Technical Specification changes (Technical Specifications 3.2.2, 4.4.1.1.1.2, 4.4.1.1.2.5, 3.4.1.3 and Figure 3.4.1.1.1-1).

1. No. The probability and consequences of accidents previously evaluated are not affected by this change. The basis for Technical Specification 2.1.1 is that boiling transition will not occur in bundles if core power is less than 25% of rated thermal power, regardless of pressure or core flow. Consequently, the specification of less than 10% rated core flow is not crucial to the basis and, thus, the use of 10 Mlbm/hr. is acceptable and has no effect on the probability or consequences of a previously evaluated accident.

For Technical Specification 2.1.2, the XN-3 critical power correlation is valid for pressure greater than or equal to 580 psig and bundle flow greater than or equal to 0.25 Mlbm/hr-ft^2. As stated in the basis for Technical Specification 2.1.1, if vessel downcomer water level is above TAF (top of active fuel), and core power greater than 25%, bundle flows for potentially limiting bundles will be greater than 0.25 Mlbm/hr-ft^2 due to natural circulation. In addition, Technical Specification 3.4.1.1.1 requires at least one (1) recirculation loop in operation to run in Condition 2, which would produce a core flow in excess of 30 Mlbm/hr. Therefore, core flows below about 30 Mlbm/hr-ft^2 are prohibited when the reactor is at power. Thus, the change from "10%" to "10 million lbm/hr" is acceptable and has no effect on the probability or consequences of a previously evaluated accident.

2. No. The basis for Technical Specification 2.1.1 is that boiling transition will not occur in bundles if core power is less than 25% of rated thermal power, regardless of pressure or core flow. The proposed change is not crucial to this basis. The XN-3 critical power correlation is valid for pressures greater than or equal to 580 psig and bundle flow greater than or equal to 0.25 Mlbm/hr-ft^2. The specification is based upon vessel downcomer water level being above TAF and core power greater than 25% which yields a bundle flow for potentially limiting bundles greater than 0.25 Mlbm/hr-ft^2 due to natural circulation. Based on Technical Specification 3.4.1.1.1, core flows below about 30 Mlbm/hr-ft^2 are prohibited when the reactor is at power. Therefore, the change to a limit of 10 Mlbm/hr is acceptable and does not create the possibility for a new or different kind of accident from any accident previously evaluated.

3. No. As explained above, the margin of safety has not been reduced.

Table 2.2.1-1 (Items 2.a, 2.b, and 2.c) and Specifications 3.2.2, 3.4.1.1.2.a.2, 3.4.1.1.2.a.3, 3.4.1.1.2.a.5.b and 3.3.6-2 (Item 2.a.1, 2.c, and 2.d), APRM Flow Biased Setpoints and Allowable Values

Although the equation for determining these setpoints does not change as a result of the power uprate, because the setpoints in these technical specifications are referenced to rated thermal power, the current limits do change in that the top portion of the operating map (power vs. reactor flow) is raised by 4.5%.

1. No. The safety analyses contained in NE-092-001 evaluated operation at both uprated power with 4.5% higher rod lines and increased core flow. In addition, General Electric Co. has analyzed and received generic approval for their BWR/4 product line operation in the Maximum Extended Operating Domain (MEOD). Operation at the 4.5% higher rod lines is bounded by the MEOD analysis. Additional justification for this small increase in the power flow operating range is contained in Section C.2.3 of NEDC-31984P.

Cycle specific reload analyses will evaluate operation at the increased power vs. flow conditions (100% uprated power vs. 87% core flow to 100% uprate power vs. 108% core flow). These analyses will ensure that the limits established in the Core Operating Limits Report are applicable to rated power operation from 87% to 108% core flow.

Based on the above analyses, increasing the current limits do not represent a significant increase in the probability or consequences of an accident previously evaluated.

2. No. The analyses described above in response to Question 1 do not indicate that a possibility for a new or different kind of accident from any previously evaluated has been created by the proposed change.

3. No. Based on the response to Question 1 above, the proposed change does not result in a reduction in the margin of safety.

Table 2.2.1-1, Item 3, Reactor Steam Dome Pressure - High Scram

The reactor steam dome pressure-high scram trip setpoint and allowable values are being changed to less than or equal to 1087 psig and less than or equal to 1093 psig respectively.

1. No. This scram function is designed to terminate a pressure increase transient not terminated by direct scram or high flux scram. The nominal trip setpoint is maintained above the reactor vessel maximum operating pressure and the specified analytical limit is used in the transient analyses. The analytical limit of 1105 psig is used in the uprated transient analyses. The results of the overpressure protection analysis indicate that the peak pressure will remain below the 1375 psig ASME limit which meets plant licensing requirements. In accordance with the methodology described in NE-092-001, transient analyses will be performed using the analytic limit and the results will be incorporated into the Core Operating Limits Report. Therefore, this proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. No. The purpose of this scram function is to terminate a pressure increase transient not terminated by direct scram or high flux scram. The nominal trip setpoint is maintained above the reactor vessel maximum operating pressure and the specified analytical limit is used in the transient analysis. 1105 psig is being used as the analytical limit in the uprated transient analysis. The results of the overpressure protection analysis indicate peak pressure will remain below the ASME limit of 1375 psig which satisfies plant licensing requirements. Based upon that result, it is concluded that the proposed change will not create the possibility of a new or different kind of accident from any previously evaluated.

3. No. The results of the overpressure protection analysis indicate peak pressure will remain below the 1375 psig licensing limit, therefore, it is concluded that the proposed change does not result in a significant reduction in a margin of safety.

Specification 4.1.5.c, Standby Liquid Control System

This specification has been revised to require SLC [Standby Liquid Control] pumps to develop a discharge pressure of greater than or equal to 1224 psig.

1. No. The ability of the SLC system to achieve and maintain safe shutdown is a function of the amount of fuel in the core and is not directly affected by core thermal power. The SLC pump test discharge pressure acceptance criteria are based on the lowest relief valve setpoint. The lowest setpoint is being increased by 30 psi (to 1106) due to power uprate. Operating with increased core flow will result in additional friction losses through the core and a slightly larger core differential pressure (approximately 4 psi). Therefore, increasing the SLC pump test discharge pressure acceptance criteria ensures the capability of SLC injection. The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. No. The ability of the SLC system to achieve and maintain safe shutdown is a function of the amount of fuel in the core and is not directly affected by core thermal power. Therefore, the proposed change does not result in a new or different kind of accident from any previously evaluated.

3. No. The ability of the SLC system to achieve and maintain safe shutdown is a function of the amount of fuel in the core and is not directly affected by core thermal power. As stated in the response to question 1 above, the SLC pump discharge pressure acceptance criteria are based upon the lowest relief valve setpoint. The lowest setpoint is being increased by 30 psi. As the SLC pumps are positive displacement pumps, the uprate will not adversely affect the performance of the pumps to achieve proper injection. Based on above, the proposed change does not result in a significant reduction in a margin of safety.

Specifications 3.2.2, 4.4.1.1.1.2, 4.4.1.1.2.5, 3.4.1.3 and Figure 3.4.1.1.1-1, Rated Core Flow References

Technical Specification 3.2.2 contains the definition of "W" for the flow biased APRM scram equation. The word "rated" is being deleted from the definition of "W" since rated core flow is being increased. The definition of "W" is not altered. The change is being made for editorial purposes.

Technical Specifications 4.4.1.1.1.1.2, 4.1.1.1.2.5, 3.4.1.3, and Figure 3.4.1.1.1-1 specify performance requirements and limits for the Reactor Recirculation System. These specifications are referenced to the current rated core flow. The references to "rated core flow" are being replaced with actual equivalent core flows. The specifications are equivalent and unchanged. This change is being made for editorial purposes to avoid confusion since rated core flow is being increased. These changes are also consistent with the changes made in Section 2.1.

1. No. The proposed changes are editorial and do not effect the probability or consequences of an accident previously evaluated.

2. No. The proposed changes are editorial and do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. No. The proposed changes are editorial and do not involve a significant reduction in a margin of safety.

Specification Table 3.3.1-1, Note (j) and Action 6, Reactor Protection System Instrumentation, and Table 3.3.4.2-1, Note b, End-of-Cycle Recirculation Pump Trip System Instrumentation

The turbine first stage pressure scram bypass at 30% power in Technical Specification Table 3.3.1-1, Note (j) and Table 3.3.4.2-1, Note (b) is revised to indicate that the uprated equivalent allowable value of first stage turbine pressure is 136 psig. This value ensures that the analytical limit of 147.7 psig, which represented 30% rated thermal power, is not exceeded.

As currently written Note (j), Note (b) and Table 3.3.1-1, ACTION 6 are unclear and could be misinterpreted. They apply only when RPS scram functions and End-of-Cycle Recirculation Pump Trip on turbine main stop valves closure or control valve fast closure are not automatically bypassed. ACTION 6 provides no guidance in the event the bypass fails to lift when thermal power is above 30%. In the worst case, the action statement could be interpreted literally to allow full power operation with the RPS function still bypassed. Such operation would violate the licensing basis analysis for the MCPR operating limit (for the Generator Load Rejection Without Bypass transient), which takes credit for operation of the anticipatory scram on control valve fast closure at greater than 30% of rated thermal power.

1. No. The revisions to Table 3.3.1-1, ACTION 6, Table 3.3.1-1, Note (j), and Table 3.3.4-1 Note (b) clarify the current requirements; they do not change their intent.

FSAR Chapter 15 transient analyses and reload licensing analyses take credit for operation of the anticipatory scram function on turbine stop valve closure and control valve fast closure for power levels greater than 30% of rated thermal power. The proposed revision to Table 3.3.1-1, ACTION 6 provides better assurance of the availability of the anticipatory scram function, since the current specifications could be interpreted literally to allow full power operation with the RPS function bypassed.

The proposed revision to Table 3.3.1-1, Note (j) and Table 3.3.4.2-1, Note (b) does not change the operation of the RPS and EOC-RPT bypasses on turbine stop valve closure and control valve fast closure below 30% power. The turbine first stage pressure switches will still be calibrated in the same manner, and, by procedure, the reactor operator will not exceed 30% power if the trip bypass annunciator does not clear.

The setpoints for the RPS and EOC-RPT bypass functions were selected to allow sufficient operating margin to avoid scrams during low power turbine generator trips. As discussed in NEDC-31894P, Section F4.2(c) and in Section 5.1.2.8 of NEDC 31948P, this small absolute setpoint increase maintains the safety basis for the setpoint.

Based on the above, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. No. The changes proposed are clarifications and do not change specification intent. The proposed change to Table 3.3.1-1, Action 6 provides better assurance of the availability of the anticipatory scram function as the specification could currently be interpreted to allow full power operation with the RPS function bypassed. The proposed changes to Table 3.3.1-1, Note (j) and Table 3.3.4-1, Note (b) do not change the operation of the RPS and EOC-RPT bypasses on turbine stop valve closure and control valve fast closure below 30% power. Therefore, the possibility for a new or different kind of accident is not created.

3. No. The proposed changes are clarification and do not change intent. Operation of the RPS and EOC-RPT bypasses on turbine stop valve closure and control valve fast closure below 30% power is not changed. Therefore, there is no reduction in the margin of safety.

Specification Table 3.3.2-2, Item 3.d, Main Steam Line Flow Differential Pressure Setpoint

The main steam line flow high differential pressure setpoint and allowable value are revised to read trip setpoint and allowable values of 113 psid and 121 psid respectively. Footnote "***" was added to Table 3.3.2-2 to indicate that these values will be confirmed during the power uprate start-up testing. If revisions to the setpoint and allowable value are required, they will be forwarded to the Commission for approval within 90 days of completion of the test program.

1. No. The main steam line flow high differential pressure setpoint changes reflect the redefinition of rated main steam line flow that occurs with power uprate. The allowable value is maintained at the same percentage of rated steam flow as the differential pressure changes due to the increased uprate steam flow. The analytical limit of 140% of uprated steam flow is maintained for the uprated analyses. The relationship between the allowable value and the analytical limit was retained to ensure that a trip avoidance margin is maintained for the normal plant testing of MSIV's and turbine stop valves. The increase in the absolute value of the trip setpoint still provides a high assurance of isolation protection for a main steam line break accident which satisfies the original intent of the design. Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. No. The increase in the absolute value of the trip setpoint still provides a high assurance of isolation protection for the main steam line break accident which satisfies the original intent of the design and, therefore does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. No. The increase in the absolute value of the trip setpoint still provides a high assurance of isolation protection for a main steam line break accident which satisfies the original intent of the design and, therefore, does not involve a significant reduction in a margin of safety.

Specification Table 3.3.2-2, Item 4.f, Isolation Actuation Instrumentation Setpoints

The RWCU system flow-high isolation trip setpoint and allowable value are being changed. System flow is being increased by 10% to maintain reactor coolant water chemistry at a level equal to pre uprate levels. The isolation setpoint change is being made to adequately maintain operating margin between normal process values and the isolation setpoints.

1. No. The basis for the RWCU flow-high isolation is to ensure a RWCU System isolation in case of a pipe break. The high flow setpoint is set high enough to avoid spurious trips from normal operating transients but low enough to ensure an isolation during a pipe break. The proposed Technical Specification limits will result in a negligible reduction in the margin between the RWCU isolation setpoint and the 4350 gpm flow postulated during a RWCU line break and will avoid spurious isolations. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. No. As stated above, the proposed change will result in only a negligible reduction in the margin between the RWCU isolation setpoint while avoiding spurious isolation. Therefore, this change maintains the original design intent and does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. No. See 1. above.

Specification Table 3.3.2-2, Items 5.a and 6.1, Isolation Actuation Instrumentation Setpoints

The HPCI and RCIC Steam Line Flow-High Technical Specifications are being changed to account for changes in steam conditions and flows that result from operation at the uprated conditions. The setpoint and allowable value for HPCI Steam Line Flow-High isolation are less than or equal to 387 inches H_2O setpoint and allowable value for the RCIC Steam Line Delta Pressure-High isolation are less than or equal to 188 inches H_2O and less than or equal to 193 inches H_2O respectively.

1. No. The bases for these setpoints are contained in the General Electric Design Specification Data Sheets for the HPCI and RCIC systems. The Design Specification Data Sheets specify that the setpoint and allowable value be set so that the isolation occurs at greater than 272% normal steam flow and less than 300% steam flow. General Electric has historically seen start-up transients as high as 272% of normal steam flow. Setting the isolation above this value prevents spurious isolations and ensures availability of the system and its safety function. Setting the isolation at less than or equal to 300% of normal flow insures that the isolation will occur if a steam line should rupture.

The existing setpoints were calculated using information obtained during the recent surveillance tests. The revised setpoints and allowable values were calculated using the current system performance and adjusted for uprate conditions in accordance with additional guidance provided in General Electric Information Letter (SIL) No. 475, Revision 2, NEDC-31336, "General Electric Setpoint Methodology," and GE Letter SPU-9378, "HPCI and RCIC Steam Line Break Detection Setpoints".

Based on the above approach, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. No. The setpoint and allowable value are set so that isolation occurs at greater than 272% normal steam flow and less than 300% steam flow. Setting the isolation at less than or equal to 300% of normal flow ensures that the isolation will occur if a steam line rupture should occur. Therefore, no new events are postulated as a result of this change.

3. No. The proposed change does not involve a significant reduction in a margin of safety as the setpoint and allowable value are set to isolate at greater than 272% normal steam flow and less than 300% steam flow which are the setpoints contained in the General Electric Design Specification Data Sheets for the HPCI and RCIC systems.

*Specification Table 4.3.2.1-1, footnote "**"*

The footnote is being changed to delete reference to reactor pressure.

1. No. The original purpose of Footnote "**" to Technical Specification Table 4.3.2.1-1 was to describe the functioning of the permissive circuitry that allowed the MSIV low condenser pressure isolation to be bypassed. The original circuitry required the Mode Switch not be in Run, the Turbine Stop Valves closed, and reactor pressure to be above setpoint. In the start-up phase of the Susquehanna Units, General Electric deleted the reactor pressure setpoint input to the bypass circuitry. Therefore, this change is being made to make the footnote conform to the installed configuration. The revised footnote is the same as found in the BWR/4 Standard Technical Specifications (NUREG 1433). This change is editorial in nature and, therefore, does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. No. Based on the response to Question 1 above, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. No. Based on the response to Question 1 above, the proposed change does not involve a significant reduction in a margin of safety.

Specification Table 3.3.6-2, Item 1.a and Specification 3.4.1.1.2.a.5.a, Rod Block Monitor Flow Biased Rod Blocks

The Rod Block Monitor (RBM) flow biased rod blocks are being changed as follows:

a. Technical Specification Table 3.3.6-2, Item 1.a is revised to read trip setpoint and allowable values of less than or equal to 0.63 W + 41% and less than or equal to 0.63 W + 43%, respectively.

b. Technical Specification 3.4.1.1.2.a.5.a is being revised to read trip setpoint and allowable values of less than or equal to 0.63 W + 35% and less than or equal to 0.63 W + 37%, respectively.

1. No. These Technical Specification changes do not represent a change from current limits. The change reflects the rescaling made necessary by the re-definition of rated thermal power.

The RBM flow biased rod blocks are used in the Rod Withdrawal Error (RWE) analysis. In order to maintain Critical Power Ratio (CPR) margins similar to previous Susquehanna cycles, the flow biased rod blocks were changed in terms of megawatts thermal but the change was not appreciable. The rescaling of the RBM flow biased rod block to reflect the re-definition of Rated Thermal Power maintains the same level of protection as previously provided. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. No. These changes do not represent a change from current limits but are rather a rescaling made necessary by the re-definition of rated thermal power.

3. No. These changes do not represent a change from current limits but are rather a rescaling made necessary by the re-definition of rated thermal power. The rescaling of the RBM flow biased rod block maintains the same level of protection as previously provided.

Specification Table 3.3.6-2, Item 2.a, Control Rod Block Instrumentation Setpoints

The APRM rod block upscale value has been changed to add a high flow clamp setpoint at 108% with a high flow clamped allowable value at 111%.

1. No. The addition of the high flow clamp to the flow biased APRM rod block function maintains the normal margins between the rod block and the scram power levels in the increased core flow regions. When the reactor core flow is greater than 100 million lbm/hr, the APRM clamp provides an alarm to help the operator avoid scrams while operating in the ICF region. This action does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. No. The changes maintain the normal margins between the rod block and the scram power levels in ICF regions. The clamp provides an alarm to avoid scrams in the ICF region.

3. No. The changes maintain the normal margins between the rod block and the scram power levels.

Specification Table 3.3.6-2, Item 6.a, Reactor Coolant System Recirculation Flow Upscale Rod Block Setpoint and Allowable Value Change

The reactor coolant system recirculation flow upscale rod block setpoint and allowable value are being increased to 114/125 divisions of full scale and 117/125 divisions of full scale respectively.

1. No. The Reactor Coolant System recirculation flow upscale rod block setpoint and allowable value are being increased to allow operation in the ICF region. The 114/125 divisions setpoint and 117/125 divisions allowable value, specified by General Electric, are based on BWR operating history.

The purpose of the Reactor Coolant System recirculation flow upscale rod block is to prevent rod movement when an abnormally high increase in reactor recirculation flow exists. An increase in reactor recirculation flow causes an increase in neutron flux that results in an increase in reactor power. However, this increase in neutron flux is monitored by the Neutron Monitoring System that can provide a rod block. No design basis accident or transient analysis takes credit for rod block signals initiated by the Reactor Coolant Recirculation System. Therefore, this change does not increase the probability or consequences of an accident previously evaluated.

2. No. Rod block signal initiation by the Reactor Coolant Recirculation System is not taken credit for in the mitigation of a design basis accident or in any transient analysis.3. No. Rod block signal initiation by the Reactor Coolant Recirculation System is not taken credit for in any transient analysis or in the mitigation of a design basis accident.

Specification 4.4.1.1.1.2 and 4.4.1.1.2.5 Reactor Coolant System

The reactor recirculation pump motor generator set scoop tube electrical and mechanical overspeed stop setpoints are being increased to a core flow of 109.5 million lbm/hr. and 110.5 million lbm/hr., respectively.

1. No. The reactor recirculation pump motor generator set scoop tube stops are being increased to allow operation at core flows in the ICF region of up to 108 million lbm/hr.

The electrical stop is maintained above the maximum operating core flow and below the mechanical stop. The 109.5 million lbm/hr. electrical stop setpoint, specified by General Electric, is based on BWR operating history. The electrical stop is a system design feature and is not used in any safety analyses.

The 110.5 million lbm/hr. mechanical stop setpoint is used in transient analysis to limit core flow during a recirculation pump controller failure. The 110.5 million lbm/hr. mechanical stop setpoint, specified by General Electric, is also based on BWR operating history. The cycle specific analyses, performed for power uprate, used the 110.5 million lbm/hr. mechanical stop setpoint.

Based on the above, this change does not involve a significant increase of the probability or consequences of an accident previously evaluated.

2. No. Increasing the reactor recirculation motor generator set scoop tube electrical and mechanical overspeed stop setpoints is being done to allow operation at core flows in the ICF region up to 108 Mlbm/hr. The electrical stop setpoint is a design feature and is not used in any safety analysis. The mechanical stop setpoint is used in transient analysis to limit core flow during a recirculation pump controller failure. Changing of this setpoint was considered in appropriate transient analyses, and will not create the possibility of a new or different kind of accident from any previously evaluated.

3. No. See 1. above. This change does not significantly reduce the margin of safety.

Specification Figure 3.4.1.1.1-1, Thermal Power Restrictions

This figure has been redrawn to reflect the new definition of Rated Thermal Power to retain the same stability operating restrictions in terms of megawatts thermal as were previously described by this graph. .

1. No. The core thermal hydraulic stability curve and associated bases are maintained at the current rod lines and power levels. Those values are redefined to reflect the redefinition of rated thermal power. Since the current operating restrictions are maintained, power uprate has no detrimental effect on the level of protection provided by these Technical Specifications. This position is consistent with NEDC-31894P, Section 5.3.3 and with NEDC-31984P, Section 3.2.

2. No. The core thermal hydraulic stability curve and associated bases are maintained at the current rod lines and power levels. Those values are changed to reflect the redefinition of rated thermal power. Since the current operating restrictions are maintained, power uprate has no detrimental effect on the level of protection provided and does not create the possibility for a new or different kind of accident.

3. No. The core thermal hydraulic stability curve and associated bases are maintained at the current rod lines and power levels. Those values are redefined to reflect the redefinition of rated thermal power. Since the current operating restrictions are maintained, there is no detrimental effect on the level of protection provided, and therefore no significant decrease in any margin of safety.

Specifications 3.4.1.1.2.5, 3.4.1.1.2.6, Reactor Coolant System, Recirculation Loops - Single Loop Operation

Specification 3.4.1.1.2.5 is being renumbered to 3.4.1.1.2.6. A new specification 3.4.1.1.2.5 is being added to specify that a 0.70 LHGR multiplier has been applied to Specification 3.2.4 when in single recirculation loop operation.

1. No. Operation with one recirculation loop out of service is allowed, but it is not considered a normal mode of operation. Single loop operation (SLO) is a special operational condition when only one of the two recirculation loops is operable. In this operating condition, the reactor power will be limited to less than 80% of rated by the maximum achievable core flow, which is typically less than 60% of rated core flow. A postulated LOCA occurring in the active recirculation loop during SLO would cause a more rapid coastdown of the recirculation flow than would occur in two loop operation, where one active loop would remain intact. This rapid coastdown causes an earlier boiling transition and deeper penetration of boiling transition into the bundle, which tends to increase the calculated PCT. However, the PCT effects of early boiling transition are substantially offset by the mitigating effect of the lower power level achievable at the start of such an event. The SAFER/GESTR-LOCA analysis results for Susquehanna for SLO and two loop operation are well below 2200°F and are documented in NEDC-32064P-1, Revision 1, "Power Uprate with Increased Core Flow Safety Analysis for Susquehanna 1 and 2", GE Nuclear Energy, July 1993.

The ECCS performance for Susquehanna under SLO was evaluated using SAFER/GESTR-LOCA. Calculations for the DBA were performed using both nominal and Appendix K inputs. The SLO SAFER/GESTR-LOCA analysis for the DBA assumes that there is essentially no period of recirculation pump coastdown. Thus, dryout is assumed to occur simultaneously at all axial locations of the hot bundle shortly after initiation of the event. Dryout is assumed to occur in one second for the nominal case and 0.1 second for the Appendix K case. These assumptions are very conservative and provide bounding results for the DBA under SLO.

The two-loop Appendix K break spectrum documented in NEDC-32064P-1 is representative of SLO because the two-loop spectrum was analyzed assuming a one second dryout time for all axial locations of the hot bundle. As shown by the two-loop break spectrum, the DBA is the limiting case for SLO. With breaks smaller than the DBA, there is a longer period of nucleate and/or film boiling prior to fuel uncovery to remove the fuel stored energy.

An LHGR multiplier of 0.70 will be imposed when the plant is in SLO. As shown in Table 5-6 of NEDC-32064P-1, the SLO results are less limiting (i.e., lower PCT's) than the results for the two loop DBA LOCA.

Thus, the licensing PCT is based appropriately on two loop operation rather than SLO.

2. No. The licensing PCT is based upon two loop operation rather than SLO, thus the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. No. Based on the response to Question 1 above, the proposed change does not

involve a significant reduction in a margin of safety.

Specification 4.4.1.1.2.3, Reactor Coolant System

Footnote **** to this Specification is being changed to reference the power uprate startup test program.

1. No. This footnote provided a mechanism for changing the power limits specified if the results of the initial startup test program determined that it was necessary. The footnote is being modified to allow operation at uprated power with the present power limits. Should the power uprate startup test program determine a need to change the power limits they will be submitted to the Commission within 90 days as required by the revised footnote. This is consistent with the original BWR startup test program philosophy and does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. No. See 1. above; this change is administrative in nature and does not create the possibility of a new or different kind of accident from any previously evaluated.

3. No. See 1. above; this change is administrative in nature and does not involve a significant reduction in a margin of safety.

Specification 3.4.2, Reactor Coolant system, Safety Relief Valves

The safety relief valve specification is being changed to reduce the number of setpoint groups from 5 to 3. Two valves will be set at 1175 psig plus or minus 1%, 6 will be set at 1195 psig plus or minus 1%. Also, the number of Operable safety valves is being increased from 10 to 12.

1. No. This change does not increase the probability of occurrence of an accident previously evaluated as, with one exception, the accidents described in FSAR Sections 5.2.2, 7.2.3, 15.1, 15.2 and 15.3 do not document any cases where the SRV's are designated as the cause or initiator of an accident. The exception is inadvertent safety relief valve opening which results in a decrease in reactor coolant inventory and/or reactor coolant temperature. The revised setpoints and proposed groupings will not increase the probability of occurrence of this type of accident.

The change does not increase the probability of occurrence of a malfunction of equipment important to safety as previously evaluated in the FSAR. The margin between peak allowable pressure and the maximum safety setpoint is unchanged. The reactor vessel and components were evaluated for the setpoint change to assure continued compliance with the structural requirements of the ASME Code. Analysis was performed on the effects of the setpoint change for the design conditions, the normal and upset conditions and the emergency and faulted conditions. The increasing RPV dome pressure does not affect the design condition and, therefore, stresses remain unchanged.

The proposed change will also not adversely affect HPCI and RCIC system performance.

There is no indication that changed setpoints contribute to an increase in probability of SRV malfunction. Reduction in the simmer margin will be compensated for by more stringent leak test requirements during valve refurbishment.

2. No. This change does not involve any hardware changes or changes in system function. Relief and safety setpoints are only slightly increased and the maximum safety setpoint remains unchanged, thus the margin between peak allowable pressure and the setpoint remains unchanged.

3. No. The technical specifications were reviewed for margins of safety applicable to the components and systems affected by the change. Analysis has been performed that demonstrates that reactor pressure will be limited to within ASME Section III allowable values for the worst case upset transient. The margin of safety is inherent in the ASME Section III allowable pressure values.

Specification 3.4.3.2.d, Reactor Coolant System, Operational Leakage

This specification is being revised to indicate that the 1 gpm leakage rate limit currently applicable applies at the uprated maximum allowable pressure of 1035 psig, plus or minus 10 psig.

1. No. The steam dome pressure for leakage is being increased by 35 psig to 1035 psig (reactor design pressure). This pressure is chosen on the basis of steam line pressure drop characteristics and excess steam flow capability of the turbine observed during plant operation up to the current rated power level. Increasing the leakage rate pressure to 1035 psig is consistent with the expected uprated operating pressure. Increasing the reactor steam dome pressure has been analyzed and found to be within allowable limits. Maintaining the leakage rate limit at 1 gpm does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. No. This change does not involve any hardware changes or change in safety function. The reactor steam dome pressure has been analyzed and found to be within allowable limits.

3. No. Maintaining leakage the rate limit at 1 gpm is conservative and does not involve a reduction in the margin of safety.

Specifications 3.4.6.2 and 4.4.6.2, Reactor Coolant System, Reactor Steam Dome

The reactor steam dome pressure limits have been changed to 1050 psig.

1. No. Operating pressure for uprated power is increased by a minimum amount necessary to assure that satisfactory reactor pressure control is maintained. The operating pressure was chosen on the basis of steam line pressure drop characteristics and excess steam flow capability of the turbine observed during plant operation up to the current rated power level. Satisfactory reactor pressure control requires an adequate flow margin between the uprated operating condition and the steam flow capability of the turbine control valves at their maximum stroke. An operating dome pressure of 1032 psig is expected and is being assumed in the transient analyses. The 1050 psig limit was chosen to maintain an adequate level of operating flexibility while maintaining an adequate distance from the high pressure scram for trip avoidance. This limit is the initial pressure value used in the overpressure protection safety analysis for power uprate, for which all licensing criteria have been met. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. No. Based on the response to Question 1. above, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. No. As described in 1. above, the 1050 psig limit was chosen to maintain an adequate level of operating flexibility while maintaining an adequate distance from the high pressure scram. This limit is the initial pressure value used in the over pressure protection safety analysis for power uprate, for which all licensing criteria have been met. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Specification 4.5.1.b.3, Emergency Core Cooling Systems

This specification has been revised to permit a test line pressure for the flow surveillance of greater than or equal to 1140 psig at nominal reactor operating conditions.

1. No. Currently, the HPCI pump test acceptance criteria discharge pressure is greater than or equal to 1266 psig. This is based, in part, on the lowest SRV setpoint of 1146 psig plus a 1% tolerance and line flow losses. For this test, the HPCI turbine is supplied with steam at the nominal operating reactor pressure of 920 +140/-20 psig. Therefore, the test requires the HPCI pump/turbine to produce an output that exceeds that which would be commensurate with the input conditions. Stated differently, HPCI would be required to develop a pump discharge pressure associated with a steam dome pressure of 1187 psig (1175 plus or minus 1% psig), while being supplied with a steam dome pressure as low as 900 psig.

The purpose of this specification is to demonstrate that the system is capable of producing the required flow at the required pressure. The concern with this approach is that while it demonstrates the required capability by achieving the actual Technical Specification value, it requires the pump turbine to "over perform". It also reduces the margin available to compensate for normal wear and tear [that] occurs and is monitored under the ASME Section XI Pump and Valve Test Program. Power uprate will be further increasing the demand because of the increase in reactor steam dome pressure.

The intent of Surveillance 4.5.1b.3 is to demonstrate that the HPCI System will produce its design flow rate at an expected reactor pressure during a LOCA. Confirmation of the capability to achieve the required flow and pressure can be satisfactorily demonstrated without requiring the pump/turbine to "over perform". This can be done by producing the nominal operating design pressure from the pump with steam supplied to the turbine at nominal reactor operating pressure. From these conditions extrapolation via pump affinity laws will show the pump discharge pressure that would be developed at emergency reactor operation conditions (i.e. lowest SRV setpoint). This value could then be compared to the calculated value required for assuring adequate core cooling in both SSES specific and generic evaluations. The HPCI System has been evaluated and shown to be capable of achieving the required pressure and flow conditions for power uprate.

Applying the method of pump affinity laws, the new Technical Specification pump discharge pressure would become greater than or equal to 1140 psig. This value is determined based on the maximum allowable test steam dome pressure of 920 + 140 = 1060 psig, plus head losses. Through the use of pump affinity laws it has been shown by calculation that achieving a value of 1140 psig at nominal reactor operating conditions will produce the required flow and pressure during emergency conditions.

Therefore, the Technical Specification HPCI pump discharge pressure at power uprate conditions is changed to greater than or equal to 1140 psig.

2. No. The methodology and the supporting change described above in the response to Question 1 above do not alter the function nor the operation of the HPCI system. Therefore, they do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. No. The methodology and the supporting change described above in response to Question 1 do not involve a significant reduction in a margin of safety.

Specification 5.4.2, Design Features, Reactor Coolant System, Volume

This specification is being changed to show that the nominal T_{ave} is being changed from 528°F to 532°F. This change is being made to reflect the higher average saturation temperature that results from a 30 psi increase in reactor design pressure.

1. No. The effects of power uprate have been evaluated to ensure that the increase in system temperatures causes minor increases in thermal loadings on pipe supports, equipment nozzles, and in-line components. The results of analyses show that at uprated conditions all ASME components will satisfy design specification requirements and code limits when evaluated to the rules of Subsection NB-3600 of the ASME Boiler and Pressure Vessel Code Section III. The effects of thermal expansion as a result of power uprate were found to be insignificant. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. No. Increases in system temperatures as a result of power uprate have been evaluated to show that increase in thermal loadings on pipe supports, equipment nozzles and in-line components are minor. Analysis shows that at all uprated conditions all ASME components will satisfy design specification requirements and code limits when evaluated to the rules of subsection NB-3600 of Section IV to the Boiler and Pressure Vessel Code. The effects of power uprate with respect to thermal expansion were found to be insignificant and, therefore, not found to create the possibility of a new or different kind of accident.

3. No. As stated above, the effects of thermal expansion as a result of power uprate were found to be insignificant. Consequently, the nominal increase in T_{ave} does not involve a significant reduction in a margin of safety.

Specification Table 5.7.1-1, Component Cyclic or Transient Limits

This specification is being changed to raise the upper limit for a heat cycle from 546°F to 551°F. This change is being made to reflect the higher average saturation temperature that results from a 30 psi increase in reactor design pressure.

1. No. The purpose of this specification is to limit the number of heatup and cooldown cycles. The effects of power uprate have been evaluated to ensure that the reactor vessel components continue to comply with the existing structural requirements of the ASME Boiler and Pressure Vessel Code. The analyses were performed for the design, normal, upset, emergency and faulted conditions. The increase in the temperature limitation is not significant with respect to the affect it has upon the RPV and associated components.

2. No. The effects of uprating power have been evaluated for the design, normal, upset, emergency and faulted conditions to ensure that the reactor vessel components continue to comply with the existing structural requirements of the ASME Boiler and Pressure Vessel Code. The increase in the temperature limitation has been found not to be significant and, therefore, does not create the possibility of a new or different kind of accident from any previously evaluated.

3. No. This specification is intended to limit the number of heatup/cooldown cycles. The increase in the temperature limitation has not been found to be significant with respect to its effects upon the RPV and its associated components and, therefore, does not significantly reduce the margin of safety.

Specification 6.9.3.2, Core Operating Limits Report

Administrative Control Section 6.9.3.2 describes and lists topical reports that are used to determine core operating limits. Topical reports 15 through 19 are LOCA methodology reports and are being deleted. These reports describe Siemens LOCA methodology. As stated in Reference 1, the GE SAFER/GESTR LOCA methodology is being used for this uprated cycle. In addition, other minor methodology changes were made for power uprate transient analysis. GE topical report NEDC-32071P, PP&L topical report NE-092-001 and the NRC Safety Evaluation Report on the PP&L power uprate licensing topical are proposed to be added as Topical Reports No. 15, 16, and 17, respectively.

1. No. These changes are editorial in nature in that only the references to documents are being changed. The methodology used to determine core limits have been previously reviewed and approved by the NRC.

2. No. See the response to Question 1 above.

3. No. See the response to Question 1 above.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room: Osterhout Free Library, Reference Department, 71 South Franklin Street, Wilkes-Barre, Pennsylvania 18701

Attorney for licensee: Jay Silberg, Esquire, Shaw, Pittman, Potts and Trowbridge, 2300 N Street NW., Washington, DC 20037

NRC Project Director: Mohan C. Thadani, Acting

Pennsylvania Power and Light Company, Docket Nos. 50-387 and 50-388 Susquehanna Steam Electric Station, Units 1 and 2, Luzerne County, Pennsylvania

Date of amendment request: July 27, 1994

Description of amendment request: This amendment will change the definition of a CORE ALTERATION included in Technical Specification Section 1.0 for each unit to allow movement and replacement of local power range monitors and control rods in a defueled cell. The new definition is consistent with the Improved Standard Technical Specifications.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration. In the submittal, the licensee stated that:

I. This proposal does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed change eliminates two previous evolutions, LPRM and Control Rod movement from a defueled cell, from being considered CORE ALTERATIONS. Thus the issue is whether the elimination of these constraints could contribute to a significant increase in the probability or consequences of a reactivity event.

Adding local power range monitors to the list of detectors which can be moved without invoking CORE ALTERATION requirements allows for the removal of these detectors for repair and replacement. Movement of these components does not impact the reactivity of the core. Therefore, allowing the movement of these detectors without invoking CORE ALTERATION provisions, does not contribute to a significant increase in the probability or consequences of a reactivity event.

Removal of a Control Rod from a defueled cell results in a negligible increase in core reactivity. Appropriate Technical Specification controls and refueling interlocks are applied during the fuel movements preceding the control rod removal to protect from or mitigate a reactivity excursion event. In addition, the design of a control rod precludes its replacement without all fuel assemblies in the cell removed. Therefore, allowing the movement of control rods from a defueled cell without invoking CORE ALTERATION provisions, does not contribute to a significant increase in the probability or consequences of a reactivity event.

The proposed Technical Specification change to adopt the revised CORE ALTERATION definition (NUREG 1433, as amended) does not effect the probability or consequences of an accident previously evaluated.

II. This proposal does not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed change eliminates two previous evolutions, LPRM and Control Rod movement from a defueled cell, from being considered CORE ALTERATIONS. Thus the issue is whether the elimination of these constraints could create the possibility of a new or different kind of accident from any accident previously evaluated.

For local power range monitors, Technical Specification 3/4.3.1 defines the minimum number of LPRMs required to be maintained operable in OPCON 5 and during Shutdown Margin Demonstration. The addition of LPRMs as an exclusion under the CORE ALTERATION definition does not change the operability requirements for the LPRMs under Technical Specification 3/4.3.1. Thus the ability of the LPRMs to perform their monitoring function is not affected by the proposed CORE ALTERATION definition change. In addition, movement of these components does not impact the reactivity of the core. Therefore, allowing the movement of these detectors without invoking CORE ALTERATION provisions, does not create the possibility of a new or different kind of accident from any accident previously evaluated.

For Control Rods, in the unlikely event that the wrong control rod was inadvertently withdrawn from a fueled cell during evolutions which were not intended to be CORE ALTERATIONS, adequate protective measures are provided by design and core monitoring instrumentation required to be operable in OPCON 5. Withdrawal of a single control rod from a cell containing fuel is bounded by Shutdown Margin analysis and demonstration. However, assuming the inadvertent control rod withdrawal resulted in a significant reactivity addition, the Reactor Protection System (RPS) would respond by inserting all control rods via the Scram function. The RPS monitors for recriticality during OPCON 5 with SRMs (except during specific controlled evolutions), IRMs, and APRMs. The Scram circuitry is completely redundant from the insert and withdrawal circuitry for the control rods. Therefore, allowing the movement of control rods from a defueled cell without invoking CORE ALTERATION provisions, does not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed Technical Specification change to adopt the revised CORE ALTERATION definition (NUREG 1433, as amended) does not create the possibility of a new or different kind of accident from any accident previously evaluated.

III. This change does not involve a significant reduction in a margin of safety.

To evaluate the potential effect on safety margin, the proposed change was evaluated as to its effect on Shutdown Margin.

Shutdown Margin defines the amount of reactivity by which the reactor is subcritical, and thus is a measure of the safety margin in avoiding unanticipated criticality events.

The movement of LPRMs does not impact the reactivity of the core, and thus does not reduce the Shutdown Margin. Removal of a Control Rod from a defueled cell results in a negligible increase in core reactivity. Therefore, the removal of a Control Rod from a defueled cell will have a negligible effect on the core Shutdown Margin. Per Technical Specification 3/4.9.10.2(c), adequate core Shutdown Margin must exist during refueling when multiple control rods and the surrounding fuel assemblies are removed from the core. Appropriate Technical Specification controls and refueling interlocks are applied during the fuel movements preceding the control rod removal to protect from or mitigate a reactivity excursion event. In addition, the core is analyzed to maintain Shutdown Margin even with the withdrawal of the highest worth rod from a fueled cell.

The proposed Technical Specification change to adopt the revised CORE ALTERATION definition (NUREG 1433, as amended) does not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

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Attorney for licensee: Jay Silberg, Esquire, Shaw, Pittman, Potts and Trowbridge, 2300 N Street NW., Washington, DC 20037

NRC Project Director: Mohan Thadani, Acting

Philadelphia Electric Company, Docket Nos. 50-352 and 50-353, Limerick Generating Station, Units 1 and 2, Montgomery County, Pennsylvania

Date of amendment request: July 20, 1994

Description of amendment request: The amendments would raise the Steam Leakage Detection system set-points that isolate the High Pressure Coolant Injection System (HPCI) and Reactor Core Isolation Cooling (RCIC) system equipment on high equipment room temperature and high delta temperature. The amendments are supported by a Limerick Generating Station modification to increase the environmental qualifications limits of the HPCI and RCIC systems to allow the systems to remain operable when equipment room cooling is unavailable.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. The proposed Technical Specifications changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

Those accident which are potentially impacted by these changes are any accident or events that require the isolation of the HPCI or RCIC system steam supply lines. This would include gross failures (pipe breaks) or significant leaks (pipe cracks) in steam lines. Minor leaks that do not significantly affect the environment in the equipment compartments are only considered with regard to being potential precursors to the development of a larger crack or break. The ability to detect small steam leaks is not dependent on the isolation instrumentation and the proposed changes to the isolation instrumentation will not impact the detection methods.

The proposed TS changes will not increase the probability of an accident since the changes will only increase the trip set-points of the instrumentation which detect increases in the temperature in the HPCI and RCIC equipment rooms. The physical establishment and setting of the proposed set-points of these accident detection and mitigation instruments will have no direct physical impact on the plant's normal operating conditions. This instrumentation is normally in a "monitoring mode," and is not actively supporting normal plant operation. Therefore, the proposed set-points can have no impact on the operating plant that would make an accident more likely to occur.

Two perspectives were evaluated regarding the potential impact on the consequences of accidents. One case is the impact on accidents which do not require HPCI or RCIC steam line isolation, but that may require the operation of the HPCI or RCIC Systems. The other case is the impact resulting from HPCI and RCIC steam line break accidents.

In the first case, the proposed changes to the set-points of these accident mitigation instruments will have no direct physical impact on the plant's accident response, except during the HPCI or RCIC pipe break accidents. During all other pipe breaks or accidents, the bounding peak HPCI and RCIC equipment compartment temperatures will still be at least 35°F below the proposed TS lower allowable values (i.e., 218°F and 198°F, respectively), and the isolation instrumentation will remain in a "monitoring mode." The isolation instrumentation will only be required to continue to passively monitor the HPCI and RCIC compartment temperatures and will meet the design basis by not inadvertently isolating the HPCI or RCIC systems.

In the second case, the HPCI and RCIC pipe break accidents described in LGS, Updated Final Safety Analysis Report (UFSAR) Section 3.6 "Protection Against Dynamic Effects Associated with the Postulated Rupture of Piping," determine the peak pressures and temperatures for the affected compartments. These peak pressures for the HPCI and RCIC breaks are the bounding pressures for breaks in these lines and, since they occur quickly, they are unaffected by the leak detection and isolation actuation systems. The peak pressures predicted in the UFSAR for the largest HPCI and RCIC steam line breaks, in the HPCI, RCIC and isolation valve compartments, are the bounding values for breaks of all sizes in these compartments. In addition, the peak temperatures are not affected by the proposed changes to the isolation actuation set-points. Therefore, the isolation of the HPCI and RCIC steam lines following a HPCI or RCIC steam line guillotine break is not dependent on the temperature trip functions, rather, the isolation is dependent on the high flow or low pressure trip functions where a delay in the response of the temperature isolation instrumentation will have no adverse impact on the consequences of the accidents described in the SAR.

An evaluation was performed to determine the potential impacts due to the proposed changes affecting the room temperatures used in the environmental qualification program. The results of this evaluation determined that the postulated peak temperatures for the HPCI pump room and the HPCI and RCIC piping areas would be at the saturation temperature for the HPCI or RCIC break blow-down in these compartments, therefore, these compartment temperatures values will not be exceeded. The RCIC pump room and isolation valve compartment environmental qualification temperatures were not postulated to be at the saturation temperature. However, this does not increase the consequences of any of the accident described in the SAR because the equipment which is normally required for RCIC system operation and which is located in the RCIC pump compartment is not required to operate following breakage of the RCIC steam supply line. The only equipment in the RCIC pump compartment that is required to operated following a RCIC steam line break is the RCIC leak detection instrumentation which are qualified to operate at temperatures greater than the saturation temperature. Finally, the isolation valve compartment postulated peak temperatures result from a HPCI steam line break in the Unit 1 and 2 isolation valve compartments. This line break produces the highest isolation valve compartment temperatures which bounds the results of a RCIC steam line break in the isolation valve compartment and the HPCI and RCIC steam lien breaks in the HPCI and RCIC pump rooms and piping areas. However, since the leak detection and isolation actuation trip set-points for the instruments in the isolation valve compartment are not being changed, then the environmental conditions in the isolation valve compartment will remain unchanged. This will assure that the isolation valves will be able to provide isolation when required.

For HPCI or RCIC leaks, the environmental conditions were not the only design basis considerations evaluated. The radiological affects were also considered. By increasing the upper allowable high ambient temperature or high delta temperature values for certain line break sizes there will be a larger total mass blow-down from the break due to the corresponding lengthening of the time to reach the higher temperature limit. However, the total integrated mass of blowdown prior to isolation of the HPCI or RCIC steam line break will still be bounded by the LGS UFSAR accident analysis and therefore, the radiological consequences of these breaks as described in the SAR will remain unchanged. These conclusions are supported by an evaluation that provided the design basis for the main steam line break and then examines the radiological consequences at the upper and lower end of the HPCI and RCIC break spectrum. Since the largest HPCI and RCIC breaks are isolated based on high flow and not based on compartment temperature increases, then the proposed changes in the temperature set-points have no impact on the radiological consequences of the design basis HPCI or RCIC pipe break accidents as described in the SAR.

The impact of the proposed changes on the probability of a malfunction of the system isolation instrumentation, valves, or the HPCI or RCIC systems was evaluated. The isolation actuation instruments are qualified for the expected environmental conditions and the proposed set-points are within the normal operating range of the instruments. Therefore, these isolation actuation instruments are more likely to randomly fail than before. In addition, by ensuring that there is no adverse impact on the ability of the HPCI or RCIC systems to respond to events which are caused by malfunctions of equipment, then the consequences of these events are not increased. An adequate margin between the proposed lower allowable trip values and the postulated equipment room environmental conditions is being maintained such that an inadvertent actuation of the HPCI or RCIC system isolation function is also no more likely to occur. The increase in the temperature isolation allowable trip values will allow increased blow-down from a pipe break or crack which will result in higher pump compartment temperatures and pressures than before for a given break size; however, the overall impact is still bounded by the LGS UFSAR Section 3.6 ruptured piping analyses. The isolation actuation instruments are qualified for the expected environmental conditions, and the proposed set-points are also within the normal operating range of the isolation instruments. Therefore, the instruments are no more likely to randomly fail and cause the loss of the HPCI or RCIC system than before. In fact, by increasing the qualification limits of the HPCI and RCIC systems, the systems will be able to remain operable with an even large steam leak in the room when room cooling is available. Therefore, the changes will have no impact on the operating plant that would increase the possibility or consequences of a malfunction of equipment important to safety.

Since the proposed changes will maintain the HPCI or RCIC steam isolation system design basis, where the consequences are bounded by an analysis contained in the LGS UFSAR, and will only change the set-points of the existing instrumentation without impacting equipment important to safety, the proposed Technical Specifications changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The proposed TS changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed TS changes will not create the possibility of a different type of accident or malfunction of equipment since the changes will only increase the trip set-points of the instrumentation which detect increases in the temperature in the HPCI and RCIC equipment rooms. The physical establishment and resetting of the set-points of these accident detection and mitigation instruments will have not direct physical impact on the plant's normal operating conditions and will not create any new accident initiators or failure modes. The severity of the potential piping system pressure transients caused by the isolation of the HPCI or RCIC steam lines at higher room temperatures remains unchanged since the isolation occurs after the postulated break blow-down has dropped to its steady state rate. Therefore, the changes will not result in a pipe break or result in any malfunction of equipment that has not previously been postulated to occur.

Therefore, the proposed set-points will not create the possibility of a different type of accident or possibility of a different type of malfunction of equipment important to safety than previously evaluated in the SAR.

3. The proposed TS changes do not involve a significant reduction in a margin of safety.

The margin of safety for the isolation actuation instrumentation as defined in the TS bases is not reduced. The proposed system isolation TS trip set-points were selected to provide equivalent margins that ensure the effectiveness of the isolation systems to mitigate the consequences of accidents without compromising the operability of the HPCI and RCIC systems. The proposed trip set-points and proposed allowable value ranges maintain adequate margins between these new values and the operating range of the HPCI and RCIC systems in order to prevent the inadvertent actuation of the isolation system and the loss of either the HPCI or RCIC systems. The differences between the trip set-points and the allowable values are being maintained as an allowance for instrument drift. The trip set-points and the allowable ranges are within the specified range of the instruments and therefore, the accuracy and drift will provide the same margin of safety as previously assumed.

Therefore, the proposed TS change do not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room: Pottstown Public Library, 500 High Street, Pottstown, Pennsylvania 19464.

Attorney for licensee: J. W. Durham, Sr., Esquire, Sr. V. P. and General Counsel, Philadelphia Electric Company, 2301 Market Street, Philadelphia, Pennsylvania 19101

NRC Project Director: Mohan C. Thadani, Acting

Philadelphia Electric Company, Docket Nos. 50-352 and 50-353, Limerick Generating Station, Units 1 and 2, Montgomery County, Pennsylvania

Date of amendment request: July 22, 1994

Description of amendment request: This amendment would remove the surveillance frequency details which govern 10 CFR 50, Appendix J, Type B and C testing from Technical Specifications.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. The proposed Technical Specifications changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed changes involve the removal of repetitious surveillance details from TS also found in 10 CFR 50, Appendix J, and rewording of TS. The removal and rewording involves no technical changes to the existing TS. The changes to the existing TS are proposed in order to be consistent with NUREG-1433. During the development of NUREG-1433, certain wording preferences or English language conventions were adopted. The proposed changes to this TS section are administrative in nature and do not impact initiators of analyzed events. They also do not impact the assumed mitigation of accidents or transient events. Therefore, the changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The proposed TS changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed changes do not involve a physical alteration of the plant or changes in methods governing normal plant operation. The proposed changes will not impose any new or different requirements or eliminate any existing requirements. Therefore, the changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed TS changes do not involve a significant reduction in a margin of safety.

The changes are administrative in nature and will not involve any technical changes. The proposed changes will not reduce a margin of safety because they have no impact on any safety analysis assumptions. In addition, because the changes are administrative in nature, no question of safety is involved. Therefore, the changes do not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room: Pottstown Public Library, 500 High Street, Pottstown, Pennsylvania 19464.

Attorney for licensee: J. W. Durham, Sr., Esquire, Sr. V. P. and General Counsel, Philadelphia Electric Company, 2301 Market Street, Philadelphia, Pennsylvania 19101

NRC Project Director: Mohan C. Thadani, Acting

Public Service Electric & Gas Company, Docket Nos. 50-272 and 50-311, Salem Nuclear Generating Station, Unit Nos. 1 and 2, Salem County, New Jersey

Date of amendment request: August 19, 1994

Description of amendment request: This change would reduce the minimum setpoints and allowable values for the Steam Generator Level - Low-Low and Low reactor protection system signals. The bases would also be modified to expand the description of the relationship between setpoints, allowable values and the plant safety analysis.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. do not involve a significant increase in the probability or consequences of an accident previously evaluated.

The Steam Generator Water Level--Low-Low signal and the Low Steam Generator Level coincident with Steam Flow/Feed Flow Mismatch signal are designed to mitigate design basis transients involving significant reductions of steam generator inventory (e.g., Loss of Normal Feedwater, Turbine Trip, Loss of Offsite Power, Feedwater Line Break). The setpoints and allowable values for these protection signals are prescribed by Technical Specifications such that performance of the signals is consistent with the plant safety analyses, considering the effects of channel uncertainties. The proposed reductions to the setpoints and allowable values for the low-low and low steam generator level signals would not affect the probability of any transient that the protection signals are designed to mitigate. The changes would reduce the probability of unnecessary reactor trips and Auxiliary Feedwater (AFW) system actuations by providing greater operating margin for plant evolutions involving steam generator level changes (e.g., plant startup). Therefore, the proposed changes do not involve any increase in probability of an accident previously evaluated.

The changes to the Steam Generator Water Level--Low-Low signal would not result in any increase in consequences of a previously analyzed accident because the proposed setpoint and allowable value would continue to ensure the safety analysis assumptions remain valid. As described in the accompanying changes to the Technical Specifications Bases, the channel uncertainty calculations performed to establish the relationships between the setpoints, allowable values and safety analyses are consistent with NRC Regulatory Guide 1.105, Revision 2. Low Steam Generator Level coincident with Steam Flow/Feed Flow Mismatch signal is not credited in the UFSAR Chapter 15 safety analyses. The proposed changes to the low steam generator level setpoint and allowable value would continue to provide reliable backup to the low-low level trip signal, consistent with IEEE-279-1971. Therefore, the proposed changes would not involve an increase in consequences of any previously analyzed accident.

2) do not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed changes would continue to ensure the appropriate reactor protection system functions (reactor trip and AFW initiation) are initiated in the event that steam generator water level decreases to the value used in the plant safety analyses. The proposed changes would not involve any changes in protection system logic or function, and do not involve any plant configurations that could adversely affect the initiation or progression of any accident sequence. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3) do not involve a significant reduction in a margin of safety.

The proposed setpoints and allowable values would continue to ensure that the assumptions in the safety analyses remain valid, with appropriate consideration of protection system channel uncertainties. Therefore, the proposed changes do not involve a reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room: Salem Free Public library, 112 West Broadway, Salem, New Jersey 08079

Attorney for licensee: Mark J. Wetterhahn, Esquire, Winston and Strawn, 1400 L Street, NW, Washington, DC 20005-3502

NRC Project Director: Mohan C. Thadani, Acting

South Carolina Electric & Gas Company, South Carolina Public Service Authority, Docket No. 50-395, Virgil C. Summer Nuclear Station, Unit No. 1, Fairfield County, South Carolina

Date of amendment request: July 20, 1994

Description of amendment request: The proposed change would modify the Virgil C. Summer Nuclear Station (VCSNS) Technical Specification (TS) Tables 2.2-1, "Reactor Trip System Instrumentation Setpoints," and 3.3-4, "Engineered Safety Features Actuation System Instrumentation Trip Setpoints," and several associated bases. The proposed change would remove three columns from the Tables. The columns contain specific rack and sensor allowable drift values.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. Operation of VCSNS in accordance with the proposed license amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

This change does not alter or delete any setpoints or Allowable Values, and as such, has no affect on any assumptions used for accident analysis. No hardware or software changes are involved, so no common mode or common cause failures can occur as a result of this change. This change has no impact on the daily operation of VCSNS. The performance of periodic calibrations and channel checks will assure the setpoints remain within tolerance. Since this amendment request affects only information that is no longer used in the daily operation of the plant and has no impact on accident analysis, the probability or consequences of an accident previously evaluated are not increased.

2. The proposed license amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

This change revises two TS tables which contain both setpoints and Allowable Values as well as other information for safety trip functions. However, the revision only deletes three columns of data that were used in determining the operability of one channel of the safety function. These values are also used in determining the setpoints and are based on measured or published tolerances and uncertainties. Although these columns are being deleted, no changes to any hardware, software, or setpoints will occur. Since these changes do not have any plant impact, no new failure mechanisms are introduced. Only the information not used on a daily basis is being removed from these tables; this will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed license amendment does not involve a significant reduction in a margin of safety.

This change revises the format of TS Tables 2.2-1 and 3.3-4 which list the setpoint and Allowable Values for safety trip functions. The data that is being removed from these tables was used to establish clear reportability requirements for any portion of one channel of any of the listed safety trip functions. Since the reporting requirements have changed and an LER is not required if one coincident channel is inoperable, this data is no longer used in daily operations. The margin of safety was established when setpoints and Allowable Values were determined, and no changes to these values are involved. There is no reduction in a margin of safety that could affect the plant, SCE&G employees, or the public.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room: Fairfield County Library, Garden and Washington Streets, Winnsboro, South Carolina 29180

Attorney for licensee: Randolph R. Mahan, South Carolina Electric & Gas Company, Post Office Box 764, Columbia, South Carolina 29218

NRC Project Director: David B. Matthews

South Carolina Electric & Gas Company, South Carolina Public Service Authority, Docket No. 50-395, Virgil C. Summer Nuclear Station, Unit No. 1, Fairfield County, South Carolina

Date of amendment request: July 20, 1994

Description of amendment request: The proposed change would modify the Virgil C. Summer Nuclear Station, Unit 1, (VCSNS) Technical Specifications (TS) to allow alternative, equivalent testing of diesel fuel used in the emergency diesel generators (EDG). These alternative methods are necessary due to recent changes in Environmental Protection Agency (EPA) regulations that are designed to limit the use of high sulfur fuels.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. The probability or consequences of an accident previously evaluated is not significantly increased.

The change in testing methods for the EDG fuel oil has no impact on the probability or consequences of any design basis accident. These tests have been determined to be equivalent to the previously approved testing methods and are needed due to changes in the EPA's regulations regarding sulfur in motor vehicle fuels. The dye used to identify high sulfur fuels will have no adverse affect on the performance of the EDG's. The proposed testing assures a continued high level of quality of the diesel fuel received and stored on site.

The change in revision level of a reference in TS section 6.9.1.11 has no impact on the probability of occurrence or consequences of any design basis accident. All design and performance criteria will continue to be met and no new single failure mechanisms will be created. The change in revision level for WCAP-10216-P-A does not involve any alterations to plant equipment or procedures which could affect any operational modes or accident precursors. This change only incorporates by reference, the methodology for determining the penalty to be used in calculating Core Operating Limits. This methodology allows the penalty to be cycle specific and is primarily affected by the core configuration. This penalty is used for normal operation and provides more conservatism to the core operation for the cycle.

2. [The proposed license amendment does not] create the possibility of a new or different kind of accident from any accident previously evaluated.

The change in testing methods for the EDG fuel oil will not create the possibility of a new or different kind of accident from any accident previously evaluated. These tests have been determined by the EPA and other organizations to be equivalent to the previously approved testing methods. The effect of the blue dye, used to identify high sulfur fuels, on the performance of the EDGs has been evaluated and determined to be insignificant. The testing proposed assures a continued high level of quality for the diesel fuel received and stored on site.

The change of revision level of a reference in TS section 6.9.1.11 has no impact on the probability of occurrence or consequences of any design basis accident. All design and performance criteria will continue to be met and no new single failure mechanisms will be created. The change in revision level for WCAP-10216-P-A does not involve any alterations to plant equipment or procedures which could affect any operational modes or accident precursors. This change only incorporates, by reference, the methodology for determining the penalty to be used in calculating Core Operating Limits. This methodology allows the penalty to be cycle specific and is primarily affected by the core configuration. This penalty is used for normal operation and provides more conservatism to the core operation for the cycle.

3. [The proposed license amendment does not] involve a significant reduction in a margin of safety.

The change in testing methods for the EDG fuel oil will not involve a significant reduction in a margin of safety. The proposed testing methods have been determined to be equivalent to the previously approved testing methods. The test for sulfur assures that the sulfur content is within the allowable range for weight-percent. The test for color and clarity assures that the fuel is relatively free of water and particulate contaminants. The proposed tests provide at least an equivalent level of quality and repeatability for the fuel oil analysis, thus assuring that the margin of safety is not reduced.

The change in revision level of a reference in TS section 6.9.1.11 does not change the proposed reload design or safety analysis limits for each cycle reload core. The associated change to WCAP-10216-P-A due to the revision will be specifically evaluated using approved reload design methods. The larger penalty actually provides for an increase in margin during certain burnup ranges. Since the safety analysis limits are unaffected, and the cycle specific analysis will show that the analysis limits are met, the change proposed will have no adverse impact on a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room: Fairfield County Library, Garden and Washington Streets, Winnsboro, South Carolina 29180

Attorney for licensee: Randolph R. Mahan, South Carolina Electric & Gas Company, Post Office Box 764, Columbia, South Carolina 29218

NRC Project Director: David B. Matthews

Tennessee Valley Authority, Docket Nos. 50-327 and 50-328, Sequoyah Nuclear Plant, Units 1 and 2, Hamilton County, Tennessee

Date of amendment request: August 19, 1994 (TS 93-09)

Description of amendment request: The proposed change would revise the implementation schedule for Amendment Nos. 182 and 174 from that stated in the amendments when they were approved by the Commission by letter dated May 24, 1994. As issued, the amendments reflected the licensee's plans to implement the changes during the Unit 2 Cycle 6 refueling outage. However, the licensee has determined that implementation would be more appropriate following the refueling outage when both units are operating in 1995. No changes to the technical specification pages other than those approved when the amendments were issued are needed.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has determined that the no significant hazards consideration exists. This analysis was provided in the original submittal for the amendment from the licensee dated October 1, 1993, and was used in the preparation of the amendments. The licensee has determined that this analysis remains valid for the proposed revision to the implementation dates and that the changes do not constitute a significant hazard. The staff previously issued the proposed finding in the Federal Register (59 FR 4947) and there were no public comments on the finding. This analysis is reproduced as follows:

TVA has evaluated the proposed technical specification (TS) change and has determined that it does not represent a significant hazards consideration based on criteria established in 10 CFR 50.92(c). Operation of Sequoyah Nuclear Plant (SQN) in accordance with the proposed amendment will not:

1. Involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed revision supports the implementation of design logic and setpoint changes to the loss-of-power relaying. This relaying is designed to ensure adequate voltage is available to safety-related loads in order to enhance their operability and support accident mitigation functions and to provide for auxiliary feedwater (AFW) pump starts. The design changes alter relay logic and delete unnecessary relaying, but do not change the diesel generator (D/G) start and load-shedding actuations that result from loss-of-power conditions. Therefore, no new actuations or functions have been created; and because the existing and proposed functions provide for accident mitigation considerations that are not the source of an accident, the probability of an accident is not increased. The deletion of the 6.9-kilovolt shutdown board normal-feeder undervoltage relays actually reduces the potential for inadvertent shutdown board blackouts as a result of short-duration voltage transients or instrument failures.

The setpoints and time delays for loss-of-power functions have been modified based on the guidelines developed by the Electrical Distribution System Clearinghouse as evaluated and determined through detailed analysis by TVA. This design is documented in TVA Calculations SQN-EEB-MS-TI06-0008, 27DAT, and DS-1-2 and is available for NRC review at the SQN site. The assigned values are conservative settings that will ensure adequate voltage is supplied to safety-related loads for accident mitigation and safety functions under normal, degraded, and loss-of-offsite-power voltage conditions with appropriate time delays to prevent damage to electrical loads and minimize premature or unnecessary actuations. The identification of loss-of-voltage conditions is enhanced by the design changes to ensure the timely sequencing of loads onto the D/G and the initiation of AFW pump starts for accident mitigation. Because there are no reductions in safety functions resulting from the design logic, setpoint, and time-delay changes to the loss-of-power instrumentation and offsite dose levels for postulated accidents will not be increased, the consequences of an accident are not increased.

The applicable mode addition, TS 3.0.4 exclusion deletion, and response time measurement clarification incorporated in the proposed change do not affect plant functions. These changes reflect the requirements that SQN has been maintaining and serve to clarify the requirements to provide consistency of application and easier understanding. The AFW footnote addition and bases revision only clarify operability conditions that are consistent with the plant design for the AFW pump and loss-of-power instrumentation. Because there are no changes to plant functions or operations, these revisions have no impact on accident probabilities or consequences.

2. Create the possibility of a new or different kind of accident from any previously analyzed.

As described above, the loss-of-power instrumentation ensures adequate voltage to safety-related loads by initiating D/G starts and load shedding and provides for AFW pump starting, but is not considered to be the source of an accident. Although the design logic, setpoint, and time-delay actuation criteria have changed, the output functions to various plant systems that actuate for load shedding and D/G starts remain the same. Therefore, actuation criteria have been affected, but not safety functions, and the TVA evaluation has confirmed that the new design enhances the ability to maintain adequate voltage to support safety functions. Since safety functions have not changed and the new loss-of-power instrumentation design continues to support operability of safety-related equipment, no new or different accident is created.

The applicable mode addition, TS 3.0.4 exclusion deletion, and response time measurement clarification, as well as the AFW operability clarifications, do not affect plant functions and will not create a new accident.

3. Involve a significant reduction in a margin of safety.

The proposed loss-of-power TS changes support design logic, setpoint, and time-delay requirements that have been verified by TVA analysis to provide acceptable voltage levels for safety-related components. In determining the acceptability of these voltage levels, the minimum voltage for operation as well as detrimental component heating resulting from sustained degraded-voltage conditions were considered. This design ensures that safety-related loads will be available and operable for normal and accident plant conditions. The applicable mode addition, TS 3.0.4 exclusion deletion, response time measurement clarification, and AFW operability clarifications provide enhancements to TS requirements and do not affect plant functions. Therefore, no safety functions are reduced by these changes and there is no reduction in the margin of safety.

The NRC has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room: Chattanooga-Hamilton County Library, 1101 Broad Street, Chattanooga, Tennessee 37402

Attorney for licensee: General Counsel, Tennessee Valley Authority, 400 West Summit Hill Drive, ET 11H, Knoxville, Tennessee 37902

NRC Project Director: Frederick J. Hebdon

Previously Published Notices Of Consideration Of Issuance Of Amendments To Facility Operating Licenses, Proposed No Significant Hazards Consideration Determination, And Opportunity For A Hearing

The following notices were previously published as separate individual notices. The notice content was the same as above. They were published as individual notices either because time did not allow the Commission to wait for this biweekly notice or because the action involved exigent circumstances. They are repeated here because the biweekly notice lists all amendments issued or proposed to be issued involving no significant hazards consideration.

For details, see the individual notice in the **Federal Register** on the day and page cited. This notice does not extend the notice period of the original notice.

Philadelphia Electric Company, Public Service Electric and Gas Company, Delmarva Power and Light Company, and Atlantic City Electric Company, Docket Nos. 50-277 and 50-278, Peach Bottom Atomic Power Station, Unit Nos. 2 and 3, York County, Pennsylvania

Date of amendment request: June 23, 1993

Brief description of amendment request: The amendments would revise the licenses and the technical specifications to change the maximum core power limit from 3293 MWt to 3458 MWt.Date of publication of individual notice in **Federal Register:** August 29, 1994 (59 FR 44432) Expiration date of individual notice: September 28, 1994

Local Public Document Room: Government Publications Section, State Library of Pennsylvania, (REGIONAL DEPOSITORY) Education Building, Walnut Street and Commonwealth Avenue, Box 1601, Harrisburg, Pennsylvania 17105.

Wisconsin Electric Power Company, Docket Nos. 50-266 and 50-301, Point Beach Nuclear Plant, Unit Nos. 1 and 2, Town of Two Creeks, Manitowoc County, Wisconsin

Date of amendments request: August 9, 1994

Brief description of amendments request: These amendments revise the Technical Specifications (TS) 5.3.4, "Steam and Power Conversion Systems," and 15.3.7, "Auxiliary Electrical Systems," to increase the allowed outage times for one motor driven auxiliary feedwater pump and for the standby emergency power for the Unit 1, Train B4160 Volt safeguards bus (A06) from 7 to 12 days. The proposed amendments would also modify TS 15.3.3, "Emergency Core Cooling System, Auxiliary Cooling Systems, Air Recirculation Fan Coolers, and Contained Spray," to provide the clarification that the service water pump (P-32E) operating with power supplied by the Alternative Shutdown System is operable from offsite power. The changes are one-time extensions of specific allowed outage times.Date of publication of individual notice in the **Federal Register:** August 19, 1994 (59 FR 42870).

Local Public Document Room: Joseph P. Mann Library, 1516 Sixteenth Street, Two Rivers, Wisconsin 54241.

Notice Of Issuance Of Amendments To Facility Operating Licenses

During the period since publication of the last biweekly notice, the Commission has issued the following amendments. The Commission has determined for each of these amendments that the application complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment.

Notice of Consideration of Issuance of Amendment to Facility Operating License, Proposed No Significant Hazards Consideration Determination, and Opportunity for A Hearing in connection with these actions was published in the **Federal Register** as indicated.

Unless otherwise indicated, the Commission has determined that these amendments satisfy the criteria for categorical exclusion in accordance with 10 CFR 51.22. Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared for these amendments. If the Commission has prepared an environmental assessment under the special circumstances provision in 10 CFR 51.12(b) and has made a determination based on that assessment, it is so indicated.

For further details with respect to the action see (1) the applications for amendment, (2) the amendment, and (3) the Commission's related letter, Safety Evaluation and/or Environmental Assessment as indicated. All of these items are available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC 20555, and at the local public document rooms for the particular facilities involved.

Baltimore Gas and Electric Company, Docket Nos. 50-317 and 50-318, Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2, Calvert County, Maryland

Date of application for amendments: November 3, 1993

Brief description of amendments: The amendments revise the Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2, Technical Specifications (TSs) by removing the TSs that are applicable to the incore instrument (ICI) system. The limitations on the use of the ICI system will be relocated to the Updated Final Safety Analysis Report. The core power distribution limits, which the ICI system is used to verify, remain in the TSs which is consistent with 10 CFR 50.36.Date of issuance: August 24, 1994Effective date: As of the date of issuance to be implemented within 30 days.

Amendment Nos.: 191 and 168

Facility Operating License Nos. DPR-53 and DPR-69: Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** December 8, 1993 (58 FR 64601) The Commission's related evaluation of these amendments is contained in a Safety Evaluation dated August 24, 1994.No significant hazards consideration comments received: No

Local Public Document Room: Calvert County Library, Prince Frederick, Maryland 20678.

Baltimore Gas and Electric Company, Docket Nos. 50-317 and 50-318, Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2, Calvert County, Maryland

Date of application for amendments: November 3, 1993

Brief description of amendments: The amendments modify the surveillance requirements to reflect the removal of the auto-closure interlock from the shutdown cooling system and revises the setpoint for the open permissive interlock.

Date of issuance: August 24, 1994

Effective date: As of the date of issuance to be implemented within 30 days.

Amendment Nos.: 192 and 169

Facility Operating License Nos. DPR-53 and DPR-69: Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** December 8, 1993 (58 FR 64600) The Commission's related evaluation of these amendments is contained in a Safety Evaluation dated August 24, 1994.No significant hazards consideration comments received: No

Local Public Document Room: Calvert County Library, Prince Frederick, Maryland 20678.

Baltimore Gas and Electric Company, Docket Nos. 50-317 and 50-318, Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2, Calvert County, Maryland

Date of application for amendments: May 27, 1994

Brief description of amendments: The amendments revise the Technical Specification surveillance test intervals from monthly to quarterly for several channel functional tests for the Reactor Protection System and the Engineered Safety Feature Actuation System. In addition, an administrative change was made to remove an out-of-date footnote concerning the Emergency Diesel Generator logic circuit modifications.

Date of issuance: August 24, 1994

Effective date: As of the date of issuance to be implemented within 30 days.

Amendment Nos.: 193 and 170

Facility Operating License Nos. DPR-53 and DPR-69: Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** July 20, 1994 (59 FR 37062) The Commission's related evaluation of these amendments is contained in a Safety Evaluation dated August 24, 1994.No significant hazards consideration comments received: No

Local Public Document Room: Calvert County Library, Prince Frederick, Maryland 20678.

Baltimore Gas and Electric Company, Docket Nos. 50-317 and 50-318, Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2, Calvert County, Maryland

Date of application for amendments: November 5, 1993, as supplemented March 11, 1994

Brief description of amendments: The amendments consist of two related changes. The first change revises the containment penetration Technical Specifications (TSs) to resemble the containment penetration TSs in NUREG-1432, "Standard Technical Specifications for Combustion Engineering Pressurized Water Reactors." The second revises the TSs to allow the containment personnel airlock to be open during fuel movement and core alterations. The TS Bases have also been revised to reflect the changes as the result of issuing these amendments.

Date of issuance: August 31, 1994

Effective date: As of the date of issuance to be implemented within 30 days.

Amendment Nos.: 194 and 171

Facility Operating License Nos. DPR-53 and DPR-69: Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** December 8, 1993 (58 FR 64602) The Commission's related evaluation of these amendments is contained in a Safety Evaluation dated August 31, 1994.No significant hazards consideration comments received: No

Local Public Document Room: Calvert County Library, Prince Frederick, Maryland 20678.

Connecticut Yankee Atomic Power Company, Docket No. 50-213, Haddam Neck Plant, Middlesex County, Connecticut

Date of application for amendment: June 16, 1994

Brief description of amendment: The amendment removes from Technical Specification 3/4.8.3, "Onsite Power Distribution," a footnote applicable for Cycle 18 only, and adds surveillance requirement 4.8.3.1.2, to test the MCC-5 automatic bus transfer feature once per refueling.

Date of Issuance: August 23, 1994

Effective date: As of the date of issuance to be implemented within 30 days.

Amendment No.: 176

Facility Operating License No. DPR-61. Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** July 20, 1994 (59 FR 37067) The Commission's related evaluation of this amendment is contained in a Safety Evaluation dated August 23, 1994.No significant hazards consideration comments received: No.

Local Public Document Room: Russell Library, 123 Broad Street, Middletown, Connecticut 06457.

Houston Lighting & Power Company, City Public Service Board of San Antonio, Central Power and Light Company, City of Austin, Texas, Docket Nos. 50-498 and 50-499, South Texas Project, Units 1 and 2, Matagorda County, Texas

Date of amendment request: April 28, 1994

Brief description of amendments: The amendments revised Technical Specification 4.6.1.3.e to add an option that will allow the personnel airlock pneumatic system leak test to be completed in 8 hours with a pressure drop of 0.50 psi.

Date of issuance: August 29, 1994

Effective date: August 29, 1994

Amendment Nos.: Unit 1 - Amendment No. 64; Unit 2 - Amendment No. 53

Facility Operating License Nos. NPF-76 and NPF-80. The amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** May 25, 1994 (59 FR 27057) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated August 29, 1994.No significant hazards consideration comments received: No

Local Public Document Room: Wharton County Junior College, J. M. Hodges Learning Center, 911 Boling Highway, Wharton, Texas 77488

Indiana Michigan Power Company, Docket Nos. 50-315 and 50-316, Donald C. Cook Nuclear Plant, Unit Nos. 1 and 2, Berrien County, Michigan

Date of application for amendments: November 15, 1993

Brief description of amendments: The amendment revises the Technical Specifications to extend the surveillance interval for the chemical analysis, inventory, and flow area of the ice condenser from 9 to 18 months.

Date of issuance: August 23, 1994

Effective date: August 23, 1994

Amendment Nos.: 180 & 164

Facility Operating License Nos. DPR-58 and DPR-74. Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** December 22, 1993 (58 FR 67849) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated August 23, 1994. No significant hazards consideration comments received: No.

Local Public Document Room: Maud Preston Palenske Memorial Library, 500 Market Street, St. Joseph, Michigan 49085.

Indiana Michigan Power Company, Docket Nos. 50-315 and 50-316, Donald C. Cook Nuclear Plant, Unit Nos. 1 and 2, Berrien County, Michigan

Date of application for amendments: September 24, 1992 and supplemented March 2, 1994.

Brief description of amendments: The amendment removes the list of containment isolation valves and associated references to the list from the Technical Specifications.

Date of issuance: August 29, 1994

Effective date: August 29, 1994

Amendment Nos.: 181 and 165

Facility Operating License Nos. DPR-58 and DPR-74. Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** February 17, 1993 (58 FR 8773) The March 2, 1994, letter provided supplemental information that was not outside the scope of this initial notice. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated August 29, 1994.No significant hazards consideration comments received: No.

Local Public Document Room: Maud Preston Palenske Memorial Library, 500 Market Street, St. Joseph, Michigan 49085.

Niagara Mohawk Power Corporation, Docket No. 50-410, Nine Mile Point Nuclear Station, Unit 2, Oswego County, New York

Date of application for amendment: July 1, 1994

Brief description of amendment: The amendment revises the secondary containment drawdown time testing requirement of Technical Specification (TS) 4.6.5.1.c.1 and the secondary containment inleakage testing requirement of TS 4.6.5.1.c.2. The amendment supports a revised design basis radiological analysis which supports an increase in secondary containment drawdown time from 6 to 60 minutes by taking credit for fission product scrubbing and retention in the suppression pool which were not assumed in the original radiological analysis but are currently assumed in the NRC's Standard Review Plan (NUREG-0800). The revised analysis also takes credit for additional mixing of primary containment and engineered safety feature system leakage with 50 percent of the secondary containment free air volume prior to the release of radioactivity to the environment. The revised radiological evaluation has determined that the radiological doses remain below 10 CFR Part 100 guideline values and General Design Criterion 19 criteria.

Date of issuance: August 30, 1994

Effective date: As of the date of issuance to be implemented within 30 days.

Amendment No.: 56

Facility Operating License No. NPF-69: Amendment revises the Technical Specifications.

Date of initial notice in **Federal Register:** July 20, 1994 (59 FR 37074) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 30, 1994.No significant hazards consideration comments received: No

Local Public Document Room: Reference and Documents Department, Penfield Library, State University of New York, Oswego, New York 13126.

Northern States Power Company, Docket No. 50-263, Monticello Nuclear Generating Plant, Wright County, Minnesota

Dates of application for amendment: November 30, 1993 and June 30, 1994.

Brief description of amendment: The proposed amendment would delete the requirements for a chlorine detection system from the following sections of Technical Specifications: 3.2.I, 3.17.A, 4.17.A, tables 4.2.1 and Technical Bases 3.2 and 3.17.A. Due to design changes at the Monticello Nuclear Generating Plant, chlorine is no longer stored onsite as a liquified gas and regulations requiring early warning of an onsite chlorine release do not apply.

Date of issuance: August 25, 1994

Effective date: August 25, 1994

Amendment No.: 89

Facility Operating License No. DPR-22. Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** March 2, 1994 (59 FR 10010) The June 30, 1994, letter provided documents cited in the amendment application and did not affect the staff's initial no significant hazards determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 25, 1994.No significant hazards consideration comments received: No.

Local Public Document Room: Minneapolis Public Library, Technology and Science Department, 300 Nicollet Mall, Minneapolis, Minnesota 55401.

Omaha Public Power District, Docket No. 50-285, Fort Calhoun Station, Unit No. 1, Washington County, Nebraska

Date of amendment request: May 21, 1993, as supplemented by letters dated September 10, 1993, and May 25, 1994

Brief description of amendment: The amendment changed the Technical specifications to reflect the relocation of the old 10 CFR 20.106 requirements to the new 10 CFR 20.1302, and to implement administrative changes.

Date of issuance: August 24, 1994

Effective date: August 24, 1994

Amendment No.: 164

Facility Operating License No. DPR-40. Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** July 7, 1993 (58 FR 36442) The additional information contained in the supplemental letters dated September 10, 1993, and May 25, 1994, was clarifying in nature and thus, within the scope of the initial notice and did not affect the staff's proposed no significant hazards consideration determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 24, 1994.No significant hazards consideration comments received: No.

Local Public Document Room: W. Dale Clark Library, 215 South 15th Street, Omaha, Nebraska 68102

Omaha Public Power District, Docket No. 50-285, Fort Calhoun Station, Unit No. 1, Washington County, Nebraska

Date of amendment request: February 12, 1993, as supplemented by letters dated August 20, 1993, and June 6, 1994

Brief description of amendment: This amendment revised Technical Secification 2.1.4, "Reactor Coolant System Leakage Limits," to implement the reactor coolant system leak-before-break methodology detection criteria. Additionally, administrative changes were made.

Date of issuance: August 25, 1994

Effective date: August 25, 1994

Amendment No.: 165

Facility Operating License No. DPR-40. Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** July 20, 1994 (59 FR 37076) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 25, 1994.No significant hazards consideration comments received: No.

Local Public Document Room: W. Dale Clark Library, 215 South 15th Street, Omaha, Nebraska 68102.

Pacific Gas and Electric Company, Docket Nos. 50-275 and 50-323, Diablo Canyon Nuclear Power Plant, Unit Nos. 1 and 2, San Luis Obispo County, California

Date of application for amendments: December 8, 1993 (Ref. LAR 93-07)

Brief description of amendments: The amendments revise the combined Technical Specifications (TS) for the Diablo Canyon Power Plant Unit Nos. 1 and 2 to revise TS 3/4.8.1, "A.C. Sources" to increase the required quantity of emergency diesel generator (EDG) fuel oil stored in the engine-mounted tank (day tank) from 200 gallons to 250 gallons. The amendment also revises TS 3/4.7.11, "Area Temperature Monitoring," and 3/4.8.1 to remove references to a five EDG configuration, based on the installation of a sixth EDG.

Date of issuance: August 23, 1994

Effective date: August 23, 1994

Amendment Nos.: 93 and 92

Facility Operating License Nos. DPR-80 and DPR-82: The amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** February 16, 1994 (59 FR

7694) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated August 23, 1994.No significant hazards consideration comments received: No.

Local Public Document Room: California Polytechnic State University, Robert E. Kennedy Library, Government Documents and Maps Department, San Luis Obispo, California 93407.

Philadelphia Electric Company, Docket No. 50-352, Limerick Generating Station, Unit 1, Montgomery County, Pennsylvania

Date of application for amendment: June 6, 1994

Brief description of amendment: This amendment removes the controls for a remote shtudown system control valve and deletes the isolation signal for certain primary containment isolation valves from TS Tables 3.3.7.4-1 and 3.6.3-1 respectively, as a result of eliminating the steam condensing mode of the Residual Heat Removal system.

Date of issuance: August 23, 1994

Effective date: August 23, 1994

Amendment Nos. 74

Facility Operating License No. NPF-39: The amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** July 20, 1994 (59 FR 37076) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated August 23, 1994. No significant hazards consideration comments received: No

Local Public Document Room: Pottstown Public Library, 500 High Street, Pottstown, Pennsylvania 19464.

Philadelphia Electric Company, Docket Nos. 50-352 and 50-353, Limerick Generating Station, Units 1 and 2, Montgomery County, Pennsylvania

Date of application for amendments: January 10, 1994, as supplemented by letter dated July 20, 1994

Brief description of amendments: The amendments relocate the seismic monitoring instrumentation Limiting Condition for Operation, Surveillance Requirements, and associated tables and Bases contained in TS Sections 3.3.7.2 and 4.3.7.2 to the Updated Final Safety Analysis Report, Section 3.7.4.

Date of issuance: August 29, 1994

Effective date: August 29, 1994

Amendment Nos. 75 and 36

Facility Operating License Nos. NPF-39 and NPF-85. The amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** March 16, 1994 (59 FR 12364) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated August 29, 1994.No significant hazards consideration comments received: No

Local Public Document Room: Pottstown Public Library, 500 High Street, Pottstown, Pennsylvania 19464.

Philadelphia Electric Company, Public Service Electric and Gas Company Delmarva Power and Light Company, and Atlantic City Electric Company,Docket Nos. 50-277 and 50-278, Peach Bottom Atomic Power Station,Unit Nos. 2 and 3, York County, Pennsylvania

Date of application for amendments: March 28, 1994, as supplemented on June 27, 1994 and July 8, 1994

Brief description of amendments: These amendments relocate the fire protection requirements from the Technical Specifications to the Updated Final Safety Analysis Report in accordance with the guidance in Generic Letter (GL) 86-10, "Implementation of Fire Protection Requirements," and GL 88-12, "Removal of Fire Protection Requirements from Technical Specifications."

Date of issuance: August 24, 1994

Effective date: August 24, 1994

Amendments Nos.: 194 and 198

Facility Operating License Nos. DPR-44 and DPR-56: Amendments revised the Technical Specifications and the licenses.

Date of initial notice in **Federal Register:** April 28, 1994 (59 FR 22012) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated August 24, 1994.No significant hazards consideration comments received: No

Local Public Document Room: Government Publications Section, State Library of Pennsylvania, (REGIONAL DEPOSITORY) Education Building, Walnut Street and Commonwealth Avenue, Box 1601, Harrisburg, Pennsylvania 17105.

Power Authority of The State of New York, Docket No. 50-286, Indian Point Nuclear Generating Unit No. 3, Westchester County, New York

Date of application for amendment: February 3, 1994

Brief description of amendment: The licensee commenced operating on a 24-month fuel cycle, instead of the previous 18-month fuel cycle, with fuel cycle 9. Fuel cycle 9 started in August 1992; however, the facility has been shut down since February 1993 for a "Performance Improvement Outage" and a restart date has not yet been established. In order to accommodate operation on a 24-month cycle after the facility restarts, the following Engineered Safety Features (ESF) instrument calibration intervals have been extended:

(1) Reactor coolant temperature instrument channels (specified in TS Table 4.1-1)

(2) Steam generator level instrument channels (specified in TS Table 4.1-1)

(3) Containment pressure instrument channels (specified in TS Table 4.1 1)

(4) Steam line pressure instrument channels (specified in TS Table 4.1-1)

(5) Turbine first stage pressure instrument channels (specified in TS Table 4.1-1)

(6) Turbine trip low auto stop oil pressure instrument channels (specified in TS Table 4.1-1)

(7) 480V bus undervoltage and alarm relays (specified in TS Table 4.1-1)

These changes followed the guidance provided in Generic Letter 91-04, "Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle," as applicable.Additionally, the following changes were also incorporated:

(8) A limiting conditions for operation requirement for a wide range containment pressure variable was added to TS Table 3.5-5 to ensure consistency with Regulatory Guide 1.97 commitments and the IP3 Emergency Operating Procedures (EOPs).

(9) A quarterly functional test surveillance requirement for the low average temperature actuation circuits of the reactor coolant temperature channels was added to Item 4 of TS Table 4.1-1.

(10) Item 14 of TS Table 4.1-1 was expanded to specify surveillance requirements for the wide range containment pressure instrumentation channels.

(11) Item 20 to TS Table 4.1-1 was revised to clarify that both the reactor trip and the ESF actuation relay logic channels are functionally tested.

Date of issuance: September 1, 1994

Effective date: As of the date of issuance to be implemented within 30 days.

Amendment No.: 150

Facility Operating License No. DPR-64: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** March 30, 1994 (59 FR 14894) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 1, 1994.No significant hazards consideration comments received: No

Local Public Document Room: White Plains Public Library, 100 Martine Avenue, White Plains, New York 10610.

Power Authority of the State of New York, Docket No. 50-333, James A. FitzPatrick Nuclear Power Plant, Oswego County, New York

Date of application for amendment: June 17, 1993, as supplemented February 24, 1994, and June 13, 1994

Brief description of amendment: The amendment adds Section 3/4.2.J., "Remote Shutdown Capability," and associated Table 3.2-10, "Remote Shutdown Capability Instrumentation and Controls," to the Technical Specifications (TSs) to provide Limiting Conditions for Operation and surveillance requirements for the remote/alternate shutdown equipment. The amendment also adds an associated Bases section to the TSs. These additions to the TSs were based on NUREG-1433, "Standard Technical Specifications - General Electric Boiling Water Reactors (BWR/4)." Several administrative changes were also made to accommodate the additions to the TSs.

Date of issuance: August 31, 1994

Effective date: As of the date of issuance to be implemented within 30 days.

Amendment No.: 216

Facility Operating License No. DPR-59: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** August 4, 1993 (58 FR 41511) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 31, 1994.No significant hazards consideration comments received: No

Local Public Document Room: Reference and Documents Department, Penfield Library, State University of New York, Oswego, New York 13126.

Public Service Electric & Gas Company, Docket Nos. 50-272 and 50-311, Salem Nuclear Generating Station, Unit Nos. 1 and 2, Salem County, New Jersey

Date of application for amendments: March 4, 1994, as supplemented on June 14, 1994 and by phone on July 22, 1994

Brief description of amendments: These amendments modify Section 5.3.1 of the Technical Specifications (TS) to allow the use of Westinghouse Vantage+ fuel with ZIRLO cladding. The previous TS required the fuel cladding to be Zircaloy-4, which is used in the Westinghouse Standard and Vantage 5H fuel designs.

Date of issuance: August 22, 1994

Effective date: August 22, 1994

Amendment Nos. 154 and 134

Facility Operating License Nos. DPR-70 and DPR-75. These amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** March 30, 1994 (59 FR 14896) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated August 22, 1994.No significant hazards consideration comments received: No

Local Public Document Room: Salem Free Public Library, 112 West Broadway, Salem, New Jersey 08079

South Carolina Electric & Gas Company, South Carolina Public Service Authority, Docket No. 50-395, Virgil C. Summer Nuclear Station, Unit No. 1, Fairfield County, South Carolina

Date of application for amendment: December 13, 1993, as supplemented February 2, 1994, and March 11, 1994.

Brief description of amendment: The amendment changes the Technical Specifications to allow for the stórage of fuel with an enrichment not to exceed a nominal 5.0 weight percent (w/o) U-235 in the VCSNS new (fresh) and spent fuel storage racks. The changes would also allow UO_2 with a maximum nominal enrichment up to 5.0 w/o U-235 to be used as fuel in the VCSNS core.

Date of issuance: August 23, 1994

Effective date: August 23, 1994

Amendment No.: 116

Facility Operating License No. NPF-12. Amendment revises the Technical Specifications.

Date of initial notice in **Federal Register:** March 16, 1994 (59 FR 12365) The March 11, 1994, letter provided clarifying information that did not change the initial determination of no significant hazards consideration as published in the **Federal Register.** The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 23, 1994. No significant hazards consideration comments received: No

Local Public Document Room: Fairfield County Library, Garden and Washington Streets, Winnsboro, South Carolina 29180.

Vermont Yankee Nuclear Power Corporation, Docket No. 50-271, Vermont Yankee Nuclear Power Station, Vernon, Vermont

Date of application for amendment: May 20, 1994

Brief description of amendment: The proposed amendment would remove Core Spray High Sparger Instrumentation from the Vermont Yankee Technical Specifications for Emergency Core Cooling System Actuation Instrumentation. In addition, an unrelated administrative change is also made.

Date of issuance: August 22, 1994

Effective date: August 22, 1994

Amendment No.: 140

Facility Operating License No. DPR-28. Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** July 6, 1994 (59 FR 34669) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 22, 1994.No significant hazards consideration comments received: No

Local Public Document Room: Brooks Memorial Library, 224 Main Street, Brattleboro, Vermont 05301.

Virginia Electric and Power Company, et al., Docket Nos. 50-338 and 50-339, North Anna Power Station, Units No. 1 and No. 2, Louisa County, Virginia

Date of application for amendments: June 9, 1994

Brief description of amendments: These amendments revise the NA-1&2 Technical Specifications (TS) by removing the Reactor Trip System and the Engineered Safety Features Actuation System response times from the TS to station-controlled documents.

Date of issuance: August 24, 1994

Effective date: August 24, 1994

Amendment Nos.: 187 and 168

Facility Operating License Nos. NPF-4 and NPF-7. Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** July 20, 1994 (59 FR 37088) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated August 24, 1994.No significant hazards consideration comments received: No.

Local Public Document Room: The Alderman Library, Special Collections Department, University of Virginia, Charlottesville, Virginia 22903-2498.

Washington Public Power Supply System, Docket No. 50-397, Nuclear Project No. 2, Benton County, Washington

Date of application for amendment: January 6, 1994

Brief description of amendment: This amendment relocates the requirements related to seismic monitoring instrumentation from the Technical Specifications (TS) to the Final Safety Analysis Report (FSAR) and plant procedures. The existing requirements will be maintained and controlled in accordance with the requirements of 10 CFR 50.59 and TS 6.8.1.

Date of issuance: August 22, 1994

Effective date: August 22, 1994, to be implemented within 30 days of issuance

Amendment No.: 131

Facility Operating License No. NPF-21: The amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** March 30, 1994 (59 FR 14902) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 22, 1994.No significant hazards consideration comments received: No.

Local Public Document Room: Richland Public Library, 955 Northgate Street, Richland, Washington 99352.

Wisconsin Electric Power Company, Docket Nos. 50-266 and 50-301, Point Beach Nuclear Plant, Unit Nos. 1 and 2, Town of Two Creeks, Manitowoc County, Wisconsin

Date of application for amendments: September 29, 1993.

Brief description of amendments: The amendments changed the inservice test frequency of the safety injection pumps, residual heat removal pumps, and containment spray pumps from monthly to quarterly. Also, the amendments added the administration of the inservice testing program to TS 15.4.2. The amendments added requirements to verify the containment sump suction is not blocked and to verify on a monthly basis, valve alignments of the emergency core cooling system and containment cooling systems.

Date of issuance: August 25, 1994

Effective date: Date of issuance to be implemented within 45 days

Amendment Nos.: 150 and 154

Facility Operating License Nos. DPR-24 and DPR-27. Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** February 2, 1994 (59 FR 4949) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 25, 1994.No significant hazards consideration comments received: No.

Local Public Document Room: Joseph P. Mann Library, 1516 Sixteenth Street, Two Rivers, Wisconsin 54241.

Wisconsin Electric Power Company, Docket Nos. 50-266 and 50-301, Point Beach Nuclear Plant, Unit Nos. 1 and 2, Town of Two Creeks, Manitowoc County, Wisconsin

Date of application for amendments: October 6, 1992

Brief description of amendments: The amendments changed all references of rod position in the Technical Specifications to units of steps rather than inches. The amendments also changed Figure 15.3.10-1 by referencing rod position in units of steps instead of percent withdrawn. Further, the amendments revised the basis for Section 15.3.10 by clarifying the definition of "fully withdrawn" as it concerns Rod Cluster Control Assemblies, and modified the basis for Section 15.3.10 to be consistent with the above changes.

Date of issuance: August 26, 1994

Effective date: Immediately, to be implemented within 45 days.

Amendment Nos.: 151 and 155

Facility Operating License Nos. DPR-24 and DPR-27. Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** March 25, 1993 (58 FR 16234) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 26, 1994No significant hazards consideration comments received: No.

Local Public Document Room: Joseph P. Mann Library, 1516 Sixteenth Street, Two Rivers, Wisconsin 54241.

Wolf Creek Nuclear Operating Corporation, Docket No. 50-482, Wolf Creek Generating Station, Coffey County, Kansas

Date of amendment request: June 7, 1994

Brief description of amendment: The amendment revises Technical Specification Table 2.2-1, "Reactor Trip System Instrumentation Setpoints," to change the over-temperature-delta-temperature (OTDT) axial flux difference (AFD) limits to reflect the results of the Cycle 8 core maneuvering analysis.

Date of issuance: August 25, 1994

Effective date: August 25, 1994

Amendment No.: 79

Facility Operating License No. NPF-42. The amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** July 6, 1994 (59 FR 34672) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated Augusty 25, 1994.No significant hazards consideration comments received: No.

Local Public Document Room locations: Emporia State University, William Allen White Library, 1200 Commercial Street, Emporia, Kansas 66801 and Washburn University School of Law Library, Topeka, Kansas 66621.

Notice Of Issuance Of Amendments To Facility Operating Licenses And Final Determination Of No Significant Hazards Consideration And Opportunity For A Hearing (Exigent Public Announcement Or Emergency Circumstances)

During the period since publication of the last biweekly notice, the Commission has issued the following amendments. The Commission has determined for each of these amendments that the application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment.

Because of exigent or emergency circumstances associated with the date the amendment was needed, there was not time for the Commission to publish, for public comment before issuance, its usual 30-day Notice of Consideration of Issuance of Amendment, Proposed No Significant Hazards Consideration Determination, and Opportunity for a Hearing.

For exigent circumstances, the Commission has either issued a **Federal Register** notice providing opportunity for public comment or has used local media to provide notice to the public in the area surrounding a licensee's facility of the licensee's application and of the Commission's proposed determination of no significant hazards consideration. The Commission has provided a reasonable opportunity for the public to comment, using its best efforts to make available to the public means of communication for the public to respond quickly, and in the case of telephone comments, the comments have been recorded or transcribed as appropriate and the licensee has been informed of the public comments.

In circumstances where failure to act in a timely way would have resulted, for example, in derating or shutdown of a nuclear power plant or in prevention of either resumption of operation or of increase in power output up to the plant's licensed power level, the Commission may not have had an opportunity to provide for public comment on its no significant hazards consideration determination. In such case, the license amendment has been issued without opportunity for comment. If there has been some time for public comment but less than 30 days, the Commission may provide an opportunity for public comment. If comments have been requested, it is so stated. In either event, the State has been consulted by telephone whenever possible.

Under its regulations, the Commission may issue and make an amendment immediately effective, notwithstanding the pendency before it of a request for a hearing from any person, in advance of the holding and completion of any required hearing, where it has determined that no significant hazards consideration is involved.

The Commission has applied the standards of 10 CFR 50.92 and has made a final determination that the amendment involves no significant hazards consideration. The basis for this determination is contained in the documents related to this action. Accordingly, the amendments have been issued and made effective as indicated.

Unless otherwise indicated, the Commission has determined that these amendments satisfy the criteria for categorical exclusion in accordance with 10 CFR 51.22. Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared for these amendments. If the Commission has prepared an environmental assessment under the special circumstances provision in 10 CFR 51.12(b) and has made a determination based on that assessment, it is so indicated.

For further details with respect to the action see (1) the application for amendment, (2) the amendment to Facility Operating License, and (3) the Commission's related letter, Safety Evaluation and/or Environmental Assessment, as indicated. All of these items are available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC 20555, and at the local public document room for the particular facility involved.

The Commission is also offering an opportunity for a hearing with respect to the issuance of the amendment. By October 14, 1994, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's "Rules of Practice for Domestic Licensing Proceedings" in 10 CFR Part 2. Interested persons should consult a current copy of 10 CFR 2.714 which is available at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC 20555 and at the local public document room for the particular facility involved. If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

As required by 10 CFR 2.714, a petition for leave to intervene shall set forth with particularity the interest of the petitioner in the proceeding, and how that interest may be affected by the results of the proceeding. The petition should specifically explain the reasons why intervention should be permitted with particular reference to the following factors: (1) the nature of the petitioner's right under the Act to be made a party to the proceeding; (2) the nature and extent of the petitioner's property, financial, or other interest in the proceeding; and (3) the possible effect of any order which may be entered in the proceeding on the petitioner's interest. The petition should also identify the specific aspect(s) of the subject matter of the proceeding as to which petitioner wishes to intervene. Any person who has filed a petition for leave to intervene or who has been admitted as a party may amend the petition without requesting leave of the Board up to 15 days prior to the first prehearing conference scheduled in the proceeding, but such an amended petition must satisfy the specificity requirements described above.

Not later than 15 days prior to the first prehearing conference scheduled in the proceeding, a petitioner shall file a supplement to the petition to intervene which must include a list of the contentions which are sought to be litigated in the matter. Each contention must consist of a specific statement of the issue of law or fact to be raised or controverted. In addition, the petitioner shall provide a brief explanation of the bases of the contention and a concise statement of the alleged facts or expert opinion which support the contention and on which the petitioner intends to rely in proving the contention at the hearing. The petitioner must also provide references to those specific sources and documents of which the petitioner is aware and on which the petitioner intends to rely to establish those facts or expert opinion. Petitioner must provide sufficient information to show that a genuine dispute exists with the applicant on a material issue of law or fact. Contentions shall be limited to matters within the scope of the amendment under consideration. The contention must be one which, if proven, would entitle the petitioner to relief. A petitioner who fails to file such a supplement which satisfies these requirements with respect to at least one contention will not be permitted to participate as a party.

Those permitted to intervene become parties to the proceeding, subject to any limitations in the order granting leave to intervene, and have the opportunity to participate fully in the conduct of the hearing, including the opportunity to present evidence and cross-examine witnesses. Since the Commission has made a final determination that the amendment involves no significant hazards consideration, if a hearing is requested, it will not stay the effectiveness of the amendment. Any hearing held would take place while the amendment is in effect.

A request for a hearing or a petition for leave to intervene must be filed with the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555, Attention: Docketing and Services Branch, or may be delivered to the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC 20555, by the above date. Where petitions are filed during the last 10 days of the notice period, it is requested that the petitioner promptly so inform the Commission by a toll-free telephone call to Western Union at 1-(800) 248-5100 (in Missouri 1-(800) 342-6700). The Western Union operator should be given Datagram Identification Number N1023 and the following message addressed to **(Project Director)**: 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, and to the attorney for the licensee.

Nontimely filings of petitions for leave to intervene, amended petitions, supplemental petitions and/or requests for a hearing will not be entertained absent a determination by the Commission, the presiding officer or the 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).

Boston Edison Company, Docket No. 50-293, Pilgrim Nuclear Power Station, Plymouth County, Massachusetts

Date of application for amendment: June 9, 1994, as supplemented August 10, 1994

Brief description of amendment: This amendment increases the allowed out-of-service time from 7 days to 14 days for the automatic depressurization system, the high pressure coolant injection system, and the reactor core isolation cooling system. A change is also made to Section 4.5.H, "Maintenance of Filled Discharge Pipe" to reflect Amendment 149 issued September 28, 1993.

Date of issuance: August 22, 1994

Effective date: August 22, 1994

Amendment No.: 156

Facility Operating License No. DPR-35: Amendment revised the Technical Specifications.Public comments requested as to proposed no significant hazards consideration: No The Commission's related evaluation of the amendment, consultation with the State, and final determination of no significant hazards consideration are contained in a Safety Evaluation dated

Local Public Document Room: Plymouth Public Library, 11 North Street, Plymouth, Massachusetts 02360.

Attorney for licensee: W. S. Stowe, Esquire, Boston Edison Company, 800 Boylston Street, 36th Floor, Boston, Massachusetts 02199.

NRC Project Director: Walter R. Butler

Duquesne Light Company, et al., Docket No. 50-412, Beaver Valley Power Station, Unit 2, Shippingport, Pennsylvania

Date of application for amendment: August 17, 1994

Brief description of amendment: The amendment changes the Technical Specifications (TS) by revising Surveillance Requirement (SR) 4.6.2.2.d of Limiting Condition For Operation (LCO) 3.6.2.2, entitled "Containment Recirculation Spray System," by adding a new footnote number (1) pertaining to 2RSS*P21A pump performance requirements. In addition, SR 4.6.2.2.e.2 is revised by deleting the footnote, denoted by a single asterisk, which pertains to an extension to the 18-month surveillance interval for first fuel cycle.

Date of issuance: August 22, 1994

Effective date: As of the date of issuance.

Amendment No: 62

Facility Operating License No. NPF-73. Amendment revised the Technical Specifications.Public comments requested as to proposed no significant hazards consideration: No. On August 17, 1994, the staff issued enforcement discretion, which was immediately effective and remained in effect until the staff's review of this amendment was completed.

The Commission's related evaluation of the amendment, finding of emergency circumstances, consultation with the Commonwealth of Pennsylvania and final no significant hazards considerations determination are contained in a Safety Evaluation dated August 22, 1994.

Local Public Document Room: B. F. Jones Memorial Library, 663 Franklin Avenue, Aliquippa, Pennsylvania 15001.

Dated at Rockville, Maryland, this 7th day of September 1994.

For The Nuclear Regulatory Commission

Jack W. Roe,

Director, Division of Reactor Projects - III/IV, Office of Nuclear Reactor Regulation

[Doc. 94–22593 Filed 9–13–94; 8:45 am]

BILLING CODE 7590–01–F

[Docket No. 70–1257]

Finding of No Significant Impact and Notice of Opportunity for a Hearing Amendment of Materials License SNM–1227, Siemen's Power Corporation Richland, WA

The U.S. Nuclear Regulatory Commission is considering the amendment of Special Nuclear License SNM–1227 for the Siemen's Power Corporation (SPC) facility located in Richland, Washington, to authorize the release of hydrofluoric (HF) acid containing less than 3 picocuries per milliliter (pCi/ml) of uranium for unrestricted use.

Summary of the Environmental Assessment

Identification of the Proposed Action: The proposed action is to amend SPC's license to allow the sale of hydrofluoric (HF) acid containing less than 3 pCi/ml of enriched uranium for use in the metal treating and chemical compounding industries. The acid is a co-product of the dry conversion process used by SPC to convert uranium hexafluoride (UF_6) to uranium dioxide (UO_2) for the fabrication of nuclear fuel. This amendment and assessment address only the sale of the HF acid co-product.

SPC is planning a major expansion of the dry conversion process and expects to increase the generation of HF acid as a result of this future expansion. SPC will apply for an amendment to expand the dry conversion process in the near future. This expansion amendment will be the subject of a future environmental assessment.

Need for the Proposed Action: SPC is authorized to store liquid process wastes in on-site lagoons and to dispose of the treated liquid wastes via the sanitary sewer to the Richland Wastewater Treatment facility. SPC currently discharges 45–50 metric tons of fluoride annually to the sewer, generated from the currently-operating ammonium diuranate (ADU) conversion lines and the prototype dry conversion line. SPC plans to expand the dry conversion process capacity and to shut down most of the ADU conversion process. This expanded dry conversion process will generate HF acid as a co-product. Sale and reuse of the HF acid from the expanded dry conversion facility will allow SPC to reduce significantly the amount of fluoride sent to the sewer.

Environmental Impacts of the Proposed Action: SPC performed a pathway analysis to estimate the total doses to an individual resulting from the sale and reuse of the HF acid and to demonstrate that these doses will not exceed the standards for protection against radiation set forth in 10 CFR Part 20 and that they are as low as reasonably achievable.

SPC estimated radiation doses to a maximally exposed individual, identified as a worker handling the HF acid in processes, including chemical milling and passivating, and in the manufacturer of cleaning solutions. The analysis considers that HF acid is highly toxic and corrosive. Doses to members of the public will be much lower than doses to individuals working with the material in an occupational capacity. The results of the analyses demonstrate that doses to a maximally exposed individual are less than 0.4 millirem per year internal dose and less than 0.02 millirem per year external dose.

The potential for public exposure to radiation from transportation accidents was also considered. The HF acid will be transported by truck in 320-gallon tanks from the Richland facility to a buyer, following Department of Transportation regulations (49 CFR Parts 173 and 178) for the transport of HF acid. In the event of a transportation accident involving the spill or release of the acid, fumes could be released. In that case, radiation exposures, to an individual member of the public could occur. However, the exposures would be of short duration, because of the toxicity and corrosivity of the HF acid, and would be considerably less than the worker dose estimate analyzed above. Emergency response actions would be carried out based upon the chemical hazards of the materials, not the radiological hazards.

Following start-up of the expanded dry conversion facility, approximately 90 percent of the liquid wastes currently being generated by the manufacturing facility will be eliminated as the dry conversion process is brought on-line and the ADU conversion process for UF_6 is closed down.

Conclusion: The dose assessment performed for the proposed action demonstrates that the doses received by members of the critical group and the exposed general population are well below the dose limits of 100 mrem/year and 25 mrem/year, as specified in 10

CFR Part 20 and 40 CFR Part 190, respectively, and are as low as reasonably achievable. To ensure that these dose limits are not exceeded, the staff recommends that the uranium concentration in the HF acid not exceed 3 pCi/ml in any batch of 20,000 liters.

Consultations with other agencies and interested persons have demonstrated that approval of this amendment will not violate any other federal, state, or local laws or regulations.

The staff concludes that there will be no significant environmental impact associated with the licensee's sale of the co-product HF acid.

Alternatives to the Proposed Action: The alternative to the proposed action is for NRC not to amend the license to allow the sale of HF acid. If the amendment is not approved, SPC would not be able to sell the HF acid co-product. In that case, SPC would continue to discharge fluoride to the Richland sewer. While this would eliminate any potential risk to human health and safety, due to the trace amount of uranium in the HF acid, there would be a continued burden on the environment because of the disposal of the fluoride via the sewer, and it would delay SPC's schedule for ultimate closure of the on0site lagoons.

SPC could use alternative management methods for the HF acid, including storage, treatment, and/or disposal. However, due to its corrosive liquid nature, the acid is not suitable for disposal without treatment. If the HF acid was considered to be a waste product, it would be a dangerous waste as defined by the Washington Dangerous Waste Regulations. However, HF acid is a commercial chemical product, and the HF acid co-product from SPC's dry conversion process is suitable for reuse as a substitute for virgin HF acid.

Agencies and Persons Consulted:

—Washington Department of Ecology, Nuclear and Mixed Waste Programs, Water Quality Section, and Shorelands Program
—Washington Department of Health, Division of Radiation Protection
—Washington Department of Fish and Wildlife
—U.S. Environmental Protection Agency, Region X
—Benton Franklin Counties Clean Air Authority
—City of Richland, Department of Water and Waste Utilities
—Yakima Indian Nation

Other sources used in the preparation of the EA include the following:

1. Amendment application and supplement from Siemens Power Corporation dated June 28 and July 7, 1994, respectively.

2. 10 CFR Part 70, Domestic Licensing of Special Nuclear Material.

3. 10 CFR Part 51, Environmental Protection Regulations for Domestic Licensing and Related Regulatory Functions.

4. 10 CFR Part 20, Standards for Protection Against Radiation.

NIOSH Pocket Guide to Chemical Hazards, U.S. Department of Health and Human Services, Public Health Service, Centers for Disease Control, National Institute for Occupational Safety and Health, 1985.

6. Threshold Limit Values and Biological Exposure Indices for 1989–1990, American Conference of Governmental Industrial Hygienists, 1989.

7. 49 CFR Part 173, Shippers—General Requirements for Shipments and Packaging.

8. 49 CFR Part 178, Specifications for Packaging.

9. Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion, U.S. EPA Federal Guidance Report No. 11, 1988.

Finding of No Significant Impact: The NRC has prepared an Environmental Assessment related to the amendment of Special Nuclear Material License SNM–1227 to allow the sale of HF acid meeting the 3 pCi/ml limit. On the basis of this assessment, NRC has concluded that environmental impacts that would result from the proposed licensing action would not be significant and do not warrant the preparation of an Environmentally Impact Statement. Accordingly, it has been determined that a Finding of No Significant Impact is appropriate.

The Environmental Assessment and the documents related to this proposed action are available for public inspection and copying at NRC's Public Document Room at the Gelman Building, 2120 L Street NW, Washington, DC.

Opportunity for a Hearing: Any person whose interest may be affected by the issuance of this amendment may file a request for a hearing. Any request for a hearing must be filed with the Office of the Secretary, U.S. Nuclear Regulatory Commission, Washington, DC 20555, within 30 days of the publication of this notice in the **Federal Register**; be served on the NRC Staff (Executive Director for Operations, One White Flint North, 11555 Rockville Pike, Rockville, MD 20852–0130) and on the licensee (Siemens Power Corporation, 2101 Horn Rapids Road, Richland, Washington, 99352–0130); and must comply with the requirements for requesting a hearing set forth in NRC's regulation, 10 CFR Part 2, Subpart L, "Informal Hearing Procedures for Adjudications in Materials Licensing Proceedings."

These requirements, which the requestor must address in detail, are:

1. The interest of the requestor in the proceeding;

2. How that interest may be affected by the results of the proceeding, including why the requestor should be permitted a hearing;

3. The requestor's area of concern about the licensing activity that is the subject matter of the proceeding; and

4. The circumstances establishing that the request for hearing is timely, that is, filed with 30 days of the date of this notice.

In addressing how the requestor's interest may be affected by the proceeding, the request should describe the nature of the requestor's right under the Atomic Energy Act of 1954, as amended, to be made a party to the proceeding; the nature and extent of the requestor's property, financial, or other (i.e., health and safety) interest in the proceeding; and the possible effect of any order that may be entered in the proceeding upon the requestor's interest.

Dated at Rockville, Maryland, this 6th day of September 1994.

For the Nuclear Regulatory Commission.

Robert C. Pierson,

Chief, Licensing Branch, Division of Fuel Cycle, Safety and Sqfeguards, NMSS.

[FR Doc. 94–22684 Filed 9–13–94; 8:45 am]

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Form N–8b–3—File No. 270–179
Form N–8b–4—File No. 270–180
Form ADV–W, Rule 203–2—File No. 270–40

Notice is hereby given pursuant to the Paperwork Reduction Act of 1980 (44 U.S.C. 3501 *et seq.*), that the Securities and Exchange Commission ("Commission") has submitted to the Office of Management and Budget