April 11, 2001

Mr. Robert G. Byram
Senior Vice President
and Chief Nuclear Officer
PPL Susquehanna, LLC
2 North Ninth Street
Allentown, PA 18101

SUBJECT: SUSQUEHANNA STEAM ELECTRIC STATION, UNITS 1 AND 2 - ISSUANCE OF AMENDMENT RE: RELAXATION OF EXCESS FLOW CHECK VALVE SURVEILLANCE REQUIREMENTS (TAC NOS. MB0425 AND MB0427)

Dear Mr. Byram:

The Commission has issued the enclosed Amendment No. 193 to Facility Operating License No. NPF-14 and Amendment No. 168 to Facility Operating License No. NPF-22 for the Susquehanna Steam Electric Station, Units 1 and 2. These amendments consist of changes to the Technical Specifications in response to your application dated October 4, 2000, as supplemented by letters dated March 12, April 2, and April 5, 2001. These amendments revise the surveillance test requirements for excess flow check valves.

A copy of our safety evaluation is also enclosed. Notice of Issuance will be included in the Commission's Biweekly Federal Register Notice.

Sincerely,

/RA/L Burkhart for

Robert G. Schaaf, Project Manager, Section 1
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-387 and 50-388

Enclosures:
1. Amendment No. 193 to License No. NPF-14
2. Amendment No. 268 to License No. NPF-22
3. Safety Evaluation

cc w/encls: See next page

Susquehanna Steam Electric Station,
Units 1 & 2

Bryan A. Snapp, Esq
Assoc. General Counsel
PPL Services Corporation
2 North Ninth Street GENTW3
Allentown, PA 18101-1179

Rocco R. Sgarro
Supervisor-Nuclear Licensing
PPL Susquehanna, LLC
2 North Ninth Street GENA61
Allentown, PA 18101-1179

Senior Resident Inspector
U.S. Nuclear Regulatory Commission
P.O. Box 35, NUCSA4
Berwick, PA 18603-0035

Director-Bureau of Radiation Protection
Pennsylvania Department of
Environmental Protection
P.O. Box 8469
Harrisburg, PA 17105-8469

PPL Susquehanna, LLC
Nuclear Records
Attn: G. DallaPalu
2 North Ninth Street GENA62
Allentown, PA 18101-1179

Richard W. Osborne
Allegheny Electric Cooperative, Inc.
212 Locust Street
P.O. Box 1266
Harrisburg, PA 17108-1266

Regional Administrator, Region 1
U.S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406

Bryce L. Shriver
Vice President-Nuclear Site Operations
Susquehanna Steam Electric Station
PPL Susquehanna, LLC
Box 467, NUCSA4
Berwick, PA 18603-0035

Herbert D. Woodeshick
Special Office of the President
PPL Susquehanna, LLC
Rural Route 1, Box 1797
Berwick, PA 18603-0035

George T. Jones
Vice President-Nuclear
Engineering & Support
PPL Susquehanna, LLC
2 North Ninth Street, GENA61
Allentown, PA 18101-1179

Dr. Judith Johnsrud
National Energy Committee
Sierra Club
443 Orlando Avenue
State College, PA 16803

Board of Supervisors
Salem Township
P.O. Box 405
Berwick, PA 18603-0035

PPL SUSQUEHANNA, LLC

ALLEGHENY ELECTRIC COOPERATIVE, INC.

DOCKET NO. 50-387

SUSQUEHANNA STEAM ELECTRIC STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 193
License No. NPF-14

1. The Nuclear Regulatory Commission (the Commission or the NRC) having found that:

   A. The application for the amendment filed by PPL Susquehanna, LLC, dated October 4, 2000, as supplemented by letters dated March 12, April 2, and April 5, 2001, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations set forth in 10 CFR Chapter I;

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

   C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) of the Facility Operating License No. NPF-14 is hereby amended to read as follows:

   (2) Technical Specifications and Environmental Protection Plan

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 193 and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. PPL Susquehanna, LLC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance and shall be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Maitri Banerjee, Acting Chief, Section 1
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: April 11, 2001

ATTACHMENT TO LICENSE AMENDMENT NO. 193

FACILITY OPERATING LICENSE NO. NPF-14

DOCKET NO. 50-387

Replace the following page of the Appendix A Technical Specifications with the attached revised page. The revised page is identified by amendment number and contains marginal lines indicating the areas of change.

REMOVE	INSERT
3.6-14	3.6-14

PPL SUSQUEHANNA, LLC

ALLEGHENY ELECTRIC COOPERATIVE, INC.

DOCKET NO. 50-388

SUSQUEHANNA STEAM ELECTRIC STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 168
License No. NPF-22

1. The Nuclear Regulatory Commission (the Commission or the NRC) having found that:

   A. The application for the amendment filed by PPL Susquehanna, LLC, dated October 4, 2000, as supplemented by letters dated March 12, April 2, and April 5 , 2001, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations set forth in 10 CFR Chapter I;

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

   C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) of the Facility Operating License No. NPF-22 is hereby amended to read as follows:

   (2) Technical Specifications and Environmental Protection Plan

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 168 and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. PPL Susquehanna, LLC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance and shall be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Maitri Banerjee, Acting Chief, Section 1
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: April 11, 2001

ATTACHMENT TO LICENSE AMENDMENT NO. 168

FACILITY OPERATING LICENSE NO. NPF-22

DOCKET NO. 50-388

Replace the following page of the Appendix A Technical Specifications with the attached revised page. The revised page is identified by amendment number and contains marginal lines indicating the areas of change.

REMOVE	INSERT
3.6-14	3.6-14

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 193 TO FACILITY OPERATING LICENSE NO. NPF-14

AND AMENDMENT NO. 168 TO FACILITY OPERATING LICENSE NO. NPF-22

PPL SUSQUEHANNA, LLC

ALLEGHENY ELECTRIC COOPERATIVE, INC.

SUSQUEHANNA STEAM ELECTRIC STATION, UNITS 1 AND 2

DOCKET NOS. 50-387 AND 50-388

1.0 INTRODUCTION

By letter dated October 4, 2000, as supplemented March 12, April 2, and April 5, 2001, PPL Susquehanna, LLC (the licensee), submitted a request for changes to the Susquehanna Steam Electric Station (SSES), Units 1 and 2, Technical Specifications (TSs). The requested changes would revise the surveillance test requirements for excess flow check valves (EFCVs). The March 12, April 2, and April 5, 2001, letters provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

2.0 EVALUATION

2.1 Purpose and Function of EFCVs

EFCVs are installed in boiling-water reactor (BWR) instrument lines which penetrate the primary containment boundary, to limit the release of fluid in the event of an instrument line break. As discussed in Regulatory Guide 1.11, "Instrument Lines Penetrating Primary Reactor Containment," the use of EFCVs satisfies the requirements of General Design Criteria 55 and 56 for automatic isolation capability of lines penetrating containment, while maintaining a highly reliable capability to monitor important parameters inside containment. Examples of EFCV installations include reactor pressure vessel level and pressure instrumentation, main steam line flow instrumentation, recirculation pump suction pressure, and reactor core isolation cooling steam line flow instrumentation.

Currently, the SSES, Units 1 and 2, TS Surveillance Requirement (SR) 3.6.1.3.9 requires that each EFCV be tested, by actuation to check flow, on a simulated instrument line break every 24 months. The proposed change revises SR 3.6.1.3.9 to verify that a representative sample (approximately 20 percent) of reactor instrument line EFCVs actuates to the isolation position, during a simulated instrument line break signal, every 24 months, such that each EFCV will be tested at least once every 10 years (nominal).

Enclosure

2.2 Topical Report NEDO-32977-A

The basis for the licensee's request is the high degree of reliability shown by EFCV's and the low consequences of an EFCV failure. The supporting analysis for the licensee's application is based on General Electric Nuclear Energy Topical Report NEDO-32977-A, DRF B21-00658-01 “Excess Flow Check Valve Testing Relaxation" dated June 2000. This report provided: (1) an estimate of steam release frequency into the reactor building due to a break in an instrument line concurrent with an EFCV failure to close and (2) an assessment of the radiological consequences of such a release.

In Topical Report NEDO-32977-A, the BWR Owners Group (BWROG) concluded that EFCVs should be tested on a staggered group test basis on a performance-based schedule not to exceed 10 years. The conclusion was based on a risk and consequences evaluation described in the topical report. The staff reviewed the topical report and issued its evaluation on March 14, 2000. In its evaluation, the staff agreed that the test interval could be extended to as much as 10 years. In conjunction with this finding, the staff noted that each licensee that adopts the relaxed test interval program for EFCVs must have a feedback mechanism and corrective action program to ensure that good performance of EFCVs is maintained. Also, each licensee is required to perform a plant-specific radiological dose assessment, EFCV failure rate analysis, and release frequency analysis to confirm that their facility is bounded by the generic analyses of the topical report.

2.3 TSTF-334

The proposed change implements Technical Specifications Task Force (TSTF) Traveler TSTF-334, Revision 2, “Relaxed Surveillance Frequency for Excess Flow Check Valve Testing,” dated October 31, 2000 (TSTF-334). TSTF-334 was received by the staff on June 23, 1999. It proposed specific changes to the Standard TS (STS) to provide guidance and facilitate licensee applications to implement the extended EFCV test intervals proposed in the topical report. It was approved by the U.S. Nuclear Regulatory Commission (NRC) staff by W. D. Beckner's letter to A. R. Pietrangelo (Nuclear Energy Institute) of October 31, 2000. In the final revision of TSTF-334, applicability was limited to those facilities that are encompassed by the analyses performed in support of the topical report, and are subject to performance and corrective action criteria to be developed by the licensee.

3.0 EVALUATION

3.1 Proposed TS Changes

The SSES, Units 1 and 2, Primary Containment Isolation Valves, TS SR 3.6.1.3.9 currently requires a demonstration that each EFCV is operable by verifying that the reactor instrumentation line EFCVs actuate to check flow on a simulated instrument line break. The specified test frequency is 24 months. The sentence in TS SR 3.6.1.3.9 will be revised to read "Verify a representative sample of reactor instrument line EFCVs actuates to check flow on a simulated instrument line break signal." The frequency at which tests are conducted would not be changed.

Also, the associated Bases will be revised under the licensee's TS Bases Control Program to include a discussion of the basis for the test frequency and the term "representative sample." The revised basis for SR 3.6.1.3.9, provided in the licensee's submittal for information, will read:

> This SR requires a demonstration that a representative sample of reactor instrumentation line excess flow check valves (EFCV[s]) are OPERABLE by verifying that the valve actuates to check flow on a simulated instrument line break. As defined in FSAR Section 6.2.4.3.5 (Reference 4), the conditions under which an EFCV will isolate, simulated instrument line break, are at flow rates which develop a differential pressure of between 3 psid and 10 psid. This SR provides assurance that the instrumentation line EFCVs will perform its design function to check flow. No specific valve leakage limits are specified because no specific leakage limits are defined in the FSAR. The 24 month Frequency is based on the need to perform some of these Surveillances under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. The representative sample consists of an approximate equal number of EFCVs such that each EFCV is tested at least once every 10 years (nominal). The nominal 10 year interval is based on other performance-based testing programs such as Inservice Testing (snubbers) and Option B to 10 CFR 50, Appendix J. In addition, the EFCVs in the sample are representative of the various plant configurations, models, sizes and operating environments. This ensures that any potential common problem with a specific type or application of EFCV is detected at the earliest possible time. EFCV failures will be evaluated to determine if additional testing in that test interval is warranted to ensure overall reliability and that failures to isolate are very infrequent. Therefore, testing of a representative sample was concluded to be acceptable from a reliability standpoint (Reference 7).

3.2 Evaluation Criteria

The staff reviewed the licensee's application for conformance to its March 14, 2000, Safety Evaluation (SE) and the guidance of the approved TSTF-334, Revision 2. The acceptance criteria relate to, and the staff's evaluation encompassed, the following areas of concern: (1) radiological dose assessment, (2) EFCV failure rate and release frequency, (3) licensee's failure feedback mechanism and corrective action program and (4) conformance of revised TS to generic TS guidance. The licensee also provided, and the staff reviewed, the potential environmental effects.

3.3 Radiological Dose Assessment

The radiological consequences for an instrument line break evaluated by the licensee do not credit the EFCVs for isolating the break. The evaluation assumes a discharge of reactor water through an instrument line with flow resistance equivalent to a 3/8-inch orifice during the detection and cooldown sequence. The assumptions of the accident analysis do not change as a result of the licensee's proposed EFCV surveillance intervals. As a result, a failure of an EFCV is bounded by the licensee's previous analysis and is consistent with the topical report results. The radiation dose consequences for an instrument line break are not impacted by the proposed change.

3.4 EFCV Failure Rate and Release Frequency

To estimate the release frequency initiated by an instrument line break, two factors are considered: (1) the instrument line break frequency downstream of the EFCV, and (2) the probability of the EFCV failing to close.

In the topical report, EFCV reliability was evaluated based on testing experience provided by 12 different BWR plants. The composite data indicated that EFCVs are very reliable. The data represented 12,424.5 valve years of operation with a total of 11 failures noted. The EFCV composite failure rate was 1.67E-07/hour and was referenced as the “upper limit” failure rate in the topical report.

A review of the topical report data indicates that Susquehanna reported 4 failures with 144 valve years noted (Unit 2 only). This data represents the second lowest hours noted and was the highest number of failures listed for plants in the topical report. The topical report did state that the failures were a conservative result in that the valves reported failed passed a subsequent bench test. The topical report also states that failures were attributed to improper test methods which were identified as test being performed at too low reactor pressure.

The NRC staff was concerned that Susquehanna reported a larger number of EFCV failures than other plants, including those in the BWROG survey results for NEDO-32977-A. The licensee submitted additional plant-specific data for SSES Units 1 and 2 that totaled 1672 valve years of operation with 16 EFCV failures noted. The data was collected for the period from the Unit 2 spring 1991 outage to the Unit 1 spring 2000 outage. The additional data provided significantly more valve years of operation but with a larger number of EFCV failures compared to other plants included in the topical report.

In the review of EFCV performance, the licensee considered valves that failed a surveillance but subsequently passed a bench test or a retest following a modification to reduce the length of the instrumentation tubing not to have failed. The staff did not agree with this conclusion since the bench test does not reflect the original test condition and no root cause was determined for the initial failed surveillance. Based on the above, the staff considered a failure population of 16 for the time interval sampled by the licensee.

The licensee stated that until 1997 it was conservatively assumed that the problems existed with the EFCVs, and as a result, 13 of the valves were replaced; although the licensee suspected that a number of valve failures were due to low test pressure based on subsequent bench testing of a limited number of failed valves.

By 1997 - 1998 the licensee determined that the majority of valve failures experienced were due to inadequate test pressure in conjunction with long lengths of instrumentation tubing limiting the differential pressure across the valves. Testing performed during the 1998, 1999, and 2000 outages resulted in 2 additional test failures (1999 and 2000). The staff considered both of these valve failures in the staff EFCV failure total.

The SSES Units 1 and 2 EFCV data was found to be consistent in time sampled but EFCV failures were higher than those given in the topical report (16 EFCV failures, 100 valves installed per unit). Using a surveillance interval of 24 months (current plant practice), an instrument line break frequency of 3.52E-03/year for SSES, and an EFCV failure frequency of

1.44E-02/year (plant-specific value), the EFCV release frequency is estimated to be 5.07E-05/year. For a surveillance interval of 10 years, the release frequency is estimated to be 2.53E-04/year. The 10-year release frequency shows an increase of 2.02E-04/year over the 24-month value. This represents the increase in the total plant release frequency for a random break of any of the 100 SSES instrument lines with a concurrent failure of the EFCV to isolate the break.

While the staff determined that the plant-specific release frequency is not consistent with the topical report release frequency, the staff considers the estimated EFCV failure rate and increase in estimated EFCV release frequency for a 10-year surveillance interval to be sufficiently low when considered in conjunction with the licensee's planned failure feedback and corrective actions discussed below in Section 3.6 of this SE.

3.5 Environmental Effects

The operational impact of an EFCV failing to close during the rupture of an instrument line connected to the reactor pressure vessel (RPV) boundary is based on environmental effects of a steam release in the vicinity of the instrument racks. The environmental impact of the failure of instrument lines connected to the RPV pressure boundary is the released steam into the reactor building. The topical report stated that the magnitude of release through an instrument line would be within the pressure control capacity of reactor building ventilation systems and that the integrity and functional performance of secondary containment following an instrument line break would be met. The licensee's analysis confirmed that an instrument line rupture outside primary containment will not result in overpressurizing secondary containment. The separation of instrument lines and equipment in the reactor building is expected to minimize the operational impact of an instrument line break on other equipment due to jet impingement. The licensee's analysis assumes plant shutdown and cooldown occur after the line break. The licensee’s evaluation of the environmental effects on the secondary containment and equipment in the reactor building is consistent with the Updated Final Safety Analysis Report, NEDO-32977-A, and Regulatory Guide 1.11, and is, therefore, acceptable.

3.6 Failure Feedback Mechanism and Corrective Action

The staff noted that the topical report does not prescribe a specific failure feedback mechanism, but does state that each plant's corrective action program must evaluate equipment failures and establish appropriate corrective actions. The licensee stated that any future reactor instrument line EFCV failure will be evaluated in accordance with the SSES corrective action program. SSES's Maintenance Rule Program will be revised to provide a means to monitor EFCV reliability. To ensure EFCV performance with the extended test interval, a new performance criterion will be established by the licensee in accordance with the requirements of the SSES Maintenance Rule Program. The acceptance criterion for reactor instrument line EFCVs will consist of the number of test failures over a specified time interval. The licensee proposed an acceptance criteria of one failure per 24-month period. This acceptance criteria is conservative with respect to the staff's topical report SE.

With plant-specific data that is not representative of the topical report results, the NRC staff considered whether the licensee's test methodology improvements (primarily involving increased test pressure) and corrective action program provided adequate assurance that future EFCV performance would be consistent with the topical report. The staff determined that

more definitive corrective actions than initially proposed by the licensee were necessary to ensure that EFCV performance will be closely monitored and be consistent with the topical report failure rates. To address the staff's concerns regarding the estimated EFCV failure rate, the licensee committed to the following corrective actions:

1. Should a test failure occur, PPL will test an additional representative sample.

2. Should a test failure occur in the additional representative sample, PPL will test all remaining representative valves.

3. For each test failure, PPL will re-test the affected valve during the subsequent test interval. This test will be in addition to the number of tests required to be performed during that interval.

The licensee defined “test failure” as failure of the valve to check flow during the as-found test.

The NRC staff considers the licensee's program to account for potential changes in EFCV failure rates to be acceptable and satisfies TSTF-334 performance and corrective action program criteria.

3.7 Conformance of Proposed TS to Generic (TSTF-334) Guidance

The staff reviewed the proposed replacement TS pages and determined that the text of the revised SR 3.6.1.3.9 is consistent with the generic guidance.

4.0 SUMMARY

On the basis of the above staff review of the licensee's radiological dose assessment, EFCV failure rate analysis, release frequency analysis, proposed replacement TS pages, EFCV failure feedback mechanism and corrective action program, and potential environmental effects, the staff finds the proposed change to relax SSES, Units 1 and 2, EFCV surveillance frequency by allowing a representative sample of EFCVs to be tested every 24 months with all EFCVs being tested at least once every 10 years (nominal) to be consistent with TSTF-334 Generic Guidance, conforming Topical Report NEDO-32977-A and the staff’s March 14, 2000 SE; and, therefore, is acceptable.

5.0 STATE CONSULTATION

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

6.0 ENVIRONMENTAL CONSIDERATION

The amendments change surveillance requirements. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (66 FR 2021). Accordingly, the

amendments meet eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

7.0 CONCLUSION

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

Principal Contributors: R. Schaaf
C. Doutt

Date: April 11, 2001

Mr. Robert G. Byram
Senior Vice President
and Chief Nuclear Officer
PPL Susquehanna, LLC
2 North Ninth Street
Allentown, PA 18101

April 11, 2001

SUBJECT: SUSQUEHANNA STEAM ELECTRIC STATION, UNITS 1 AND 2 - ISSUANCE OF AMENDMENT RE: RELAXATION OF EXCESS FLOW CHECK VALVE SURVEILLANCE REQUIREMENTS (TAC NOS. MB0425 AND MB0427)

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A copy of our safety evaluation is also enclosed. Notice of Issuance will be included in the Commission's Biweekly Federal Register Notice.

Sincerely,
/RA/ LBurkhart for

Robert G. Schaaf, Project Manager, Section 1
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-387 and 50-388

Enclosures: 1. Amendment No. 193 to
License No. NPF-14
2. Amendment No. 22 to
License No. NPF-22
3. Safety Evaluation

cc w/encls: See next page

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PUBLIC	M. O'Brien	C. Doutt	M. Rubin
PDI-1 Reading	R. Schaaf	W. Beckner	
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M. Banerjee	G. Hill(4)	D. Florek, RGN-I	

ACCESSION NO. [C:\amdb0425.wpd]

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