ATTACHMENT 3

PROPOSED LICENSE AMENDMENTS FOR
ADMINISTRATIVE CHANGES TO TECHNICAL SPECIFICATIONS

PROPOSED TECHNICAL SPECIFICATIONS PAGES

TURKEY POINT UNITS 3 AND 4

Section 6 Marked-Up Pages

Note: For continuity and completeness of text, and for ease of review, this section also includes pages from the TS Section 6.0 which have no change along with the pages with proposed changes as marked.

ADMINISTRATIVE CONTROLS

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6.1 RESPONSIBILITY

6.1.1 The Plant General Manager shall be responsible for overall unit operation of both units and shall delegate in writing the succession to this responsibility during his absence.

6.1.2 The Nuclear Plant Supervisor (or during his absence from the control room, a designated individual) shall be responsible for the control room command function. A management directive to this effect, signed by the Site Vice President shall be reissued to all station personnel on an annual basis.

6.2 ORGANIZATION

ONSITE AND OFFSITE ORGANIZATION

6.2.1 An onsite and an offsite organization shall be established for facility operation and corporate management. The onsite and offsite organization shall include the positions for activities affecting the safety of the nuclear power plant.

a. Lines of authority, responsibility and communication shall be established and defined from the highest management levels through intermediate levels to, and including all operating organization positions. Those relationships shall be documented and updated, as appropriate, in the form of organizational charts. These organizational charts will be documented in the Topical Quality Assurance Report and updated in accordance with 10 CFR 50.54(a)(3).

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

c. The Plant General Manager shall be responsible for overall plant safe operation and shall have control over those onsite activities necessary for safe operation and maintenance of the plant.

d. Although the individuals who train the operating staff and those who carry out the quality assurance functions may report to the appropriate manager onsite, they shall have sufficient organizational freedom to be independent from operating pressures.

e. Although health physics individuals may report to any appropriate manager onsite, for matters relating to radiological health and safety of employees and the public, the Health Physics Supervisor shall have direct access to that onsite individual having responsibility for overall unit management. Health physics personnel shall have the authority to cease any work activity when worker safety is jeopardized or in the event of unnecessary personnel radiation exposures.

PLANT STAFF

6.2.2 The plant organization shall be subject to the following:

a. Each on-duty shift shall be composed of at least the minimum shift crew composition shown in Table 6.2-1;

b. At least one licensed Operator shall be in the control room when fuel is in either reactor.

c. At least two licensed Operators shall be present in the control room during reactor startup, scheduled reactor shutdown and during recovery from reactor trips. In addition, while either unit is in MODE 1, 2, 3, or 4, at least one licensed Senior Operator shall be in the control room;

d. A Health Physics Technician* shall be on site when fuel is in the reactor;

e. All CORE ALTERATIONS shall be observed and directly supervised by either a licensed Senior Operator or licensed Senior Operator Limited to Fuel Handling who has no other concurrent responsibilities during this operation; and

f. ~~Administrative procedures shall be developed and implemented to limit the working hours of plant staff who perform safety-related functions (e.g., licensed Senior Operators, licensed Operators, health physicists, auxiliary operators, and key maintenance personnel).~~

DELETED.

~~The procedures shall include guidelines on working hours that ensure that adequate shift coverage is maintained without routine heavy use of overtime for individuals. However, in the event that unforeseen problems require substantial amounts of overtime to be used, or during extended periods of shut-down for refueling, major maintenance, or major plant modification, on a temporary basis the following guidelines shall be followed:~~

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* The Health Physics Technician composition may be less than the minimum requirements for a period of time not to exceed 2 hours, in order to accommodate unexpected absence, provided immediate action is taken to fill the required positions.

PLANT STAFF (Continued)

DELETED

1. An individual should not be permitted to work more than 16 hours straight, excluding shift turnover time.

2. An individual should not be permitted to work more than 16 hours in any 24-hour period, nor more than 24 hours in any 48-hour period, nor more than 72 hours in any 7-day period, all excluding shift turnover time.

3. A break of at least 8 hours should be allowed between work periods, including shift turnover time.

4. Except during extended shutdown periods, the use of overtime should be considered on an individual basis and not for the entire staff on a shift.

Any deviation from the above guidelines shall be authorized by the applicable department manager or higher levels of management, in accordance with established procedures and with documentation of the basis for granting the deviation. Controls shall be included in the procedures such that individual overtime shall be reviewed monthly by the Plant General Manager or his designee to assure that excessive hours have not been assigned. Routine deviation from the above guidelines is not authorized.

h. The Operations Supervisor shall hold a Senior Reactor Operator License.

i. The Operations Manager shall either:

1. hold or have held a Senior Reactor Operator License on the Turkey Point Plant; or,

2. have held a Senior Reactor Operator License on a similar plant (i.e., another pressurized water reactor); or

3. have completed the Turkey Point Plant Senior Management Operations Training Course. (i.e., certified at an appropriate simulator for equivalent senior operator knowledge level.)

TABLE 6.2-1

MINIMUM SHIFT CREW COMPOSITION

POSITION	NUMBER OF INDIVIDUALS REQUIRED TO FILL POSITION		
	BOTH UNITS IN MODE 1, 2, 3, or 4	BOTH UNITS IN MODE 5 or 6 OR DEFUELED	ONE UNIT IN MODE 1, 2, 3, or 4 AND ONE UNIT IN MODE 5 or 6 or DEFUELED
NPS	1	1	1
SRO	1	none**	1
RO	3*	2*	3*
AO	3*	3*	3*
STA	1***	none	1***

NPS - Nuclear Plant Supervisor with a Senior Operator license
SRO - Individual with a Senior Operator license
RO - Individual with an Operator license
AO - Auxiliary Operator
STA - Shift Technical Advisor

The shift crew composition may be one less than the minimum requirements of Table 6.2-1 for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements of Table 6.2-1. This provision does not permit any shift crew position to be unmanned upon shift change due to an oncoming shift crewman being late or absent.

During any absence of the Nuclear Plant Supervisor from the control room while a unit is in MODE 1, 2, 3, or 4, an individual (other than the Shift Technical Advisor) with a valid Senior Operator license shall be designated to assume the control room command function. During any absence of the Nuclear Plant Supervisor from the control room while both units are in MODE 5 or 6, an individual with a valid Senior Operator license or Operator license shall be designated to assume the control room command function.

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*At least one of the required individuals must be assigned to the designated position for each unit.

**At least one licensed Senior Operator or licensed Senior Operator Limited to Fuel Handling must be present during CORE ALTERATIONS on either unit, who has no other concurrent responsibilities.

***The STA position ~~shall be manned in MODES 1, 2, 3, and 4 unless~~ may be filled by the Nuclear Plant Supervisor or ~~the~~ an individual with a Senior Operator license who meets the ~~qualifications for the STA as required by the NRC~~ 1985 NRC Policy Statement on Engineering Expertise on Shift.

6.2.3 SHIFT TECHNICAL ADVISOR

meet the qualifications specified by the 1985 NRC Policy Statement on Engineering Expertise on Shift.

6.2.3.1 The Shift Technical Advisor shall provide advisory technical support ~~to the Nuclear Plant Supervisor~~ in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit and the opposite unit. The Shift Technical Advisor shall have a bachelor's degree or equivalent in a scientific or engineering discipline and shall ~~have received specific training in the response and analysis of the unit for transients and accidents, and in unit design and layout, including the capabilities of instrumentation and controls in the control room.~~

6.3 FACILITY STAFF QUALIFICATIONS

6.3.1 Each member of the facility staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions except for

6.3.1.1 The Health Physics Supervisor who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975.

6.3.1.2 The Operations Manager whose requirement for a Senior Reactor Operator License is as stated in Specification 6.2.2.i.

6.3.1.3 The licensed Operators and Senior Operators who shall also meet or exceed the minimum qualifications of the supplemental requirements specified in 10 CFR Part 55, and ANSI 3.1, 1981.

6.3.1.4 The Multi-Discipline Supervisors who shall meet or exceed the following requirements:

a. Education: Minimum of a high school diploma or equivalent

b. Experience: Minimum of four years of related technical experience, which shall include three years power plant experience of which one year is at a nuclear power plant

c. Training: Complete the Multi-Discipline Supervisor training program

6.3.2 When the Health Physics Supervisor does not meet the above requirements, compensatory action shall be taken which the Plant Nuclear Safety Committee determines and the NRC office of Nuclear Reactor Regulation concurs that the action meets the intent of Specification 6.3.1.

DELETED

6.4 ~~TRAINING~~

6.4.1 A retraining and replacement training program for the facility staff shall be maintained under the direction of the Training Manager and shall meet or exceed the requirements and recommendations of Section 5.5 of ANSI N18.1-1971, 10 CFR Part 55 and ANSI 3.1, 1981 and shall include familiarization with relevant industry operational experience.

6.4.2 A training program for the fire brigade shall be maintained under the direction of the Fire Protection Supervisor and shall meet or exceed the requirements of 10 CFR 50.48 and 10 CFR 50 Appendix R.

6.5 ~~REVIEW AND AUDIT~~

DELETED

~~6.5.1 PLANT NUCLEAR SAFETY COMMITTEE (PNSC)~~

~~FUNCTION~~

~~6.5.1.1 The PNSC shall function to advise the Plant General Manager on all matters related to nuclear safety.~~

DELETED

COMPOSITION

6.5.1.2 The PNSC shall have a minimum of nine voting members and be composed of individuals from each of the following disciplines:

Operations
Maintenance
Health Physics
Reactor Engineering
Protection Services
Technical Support
Licensing
Quality Assurance/Control
Instrument and Control

The PNSC Chairman and Vice-Chairman shall be appointed in writing from among the members by the Plant General Manager.

The members, according to individual job titles, shall meet the requirements as described in Sections 4.2, 4.3, or 4.4, of the ANSI N-18.1-1971.

ALTERNATES

6.5.1.3 All alternate members shall be appointed in writing by the Plant General Manager to serve on a temporary basis; however, no more than two alternates shall participate as members in PNSC activities at any one time.

MEETING FREQUENCY

6.5.1.4 The PNSC shall meet at least once per calendar month and as convened by the PNSC Chairman or his designated alternate.

QUORUM

6.5.1.5 The quorum of the PNSC necessary for the performance of the PNSC responsibility and authority provisions of these Technical Specifications shall consist of the Chairman or Vice Chairman and four members including alternates.

RESPONSIBILITIES

6.5.1.6 The PNSC shall be responsible for:

a. Review of all safety-related plant administrative procedures and changes thereto.

b. Review of all proposed tests and experiments that affect nuclear safety;

c. Review of all proposed changes to Appendix "A" Technical Specifications;

d. Review of all proposed changes or modifications to unit systems or equipment that affect nuclear safety;

RESPONSIBILITIES (Continued) DELETED

e. Investigation of all violations of the Technical Specifications, including the preparation and forwarding of reports covering evaluation and recommendations to prevent recurrence, to the President-Nuclear Division and to the Chairman of the Company Nuclear Review Board;

f. Review of all REPORTABLE EVENTS;

g. Review of reports of significant operating abnormalities or deviations from normal and expected performance of plant equipment or systems that affect nuclear safety.

h. Performance of special reviews, investigations, or analyses and reports thereon as requested by the Plant General Manager or the Chairman of the Company Nuclear Review Board;

i. Review of changes to the PROCESS CONTROL PROGRAM and the OFFSITE DOSE CALCULATION MANUAL;

j. Review of any accidental, unplanned, or uncontrolled radioactive release including the preparation of reports covering evaluation, recommendations, and disposition of the corrective action to prevent recurrence and the forwarding of these reports to the President-Nuclear Division and to the Chairman of the Company Nuclear Review Board.

k. Review of the Fire Protection Program and implementing procedures and the submittal of recommended changes to the Company Nuclear Review Board.

l. Review of the Diesel Fuel Oil Testing Program and implementing procedures.

6.5.1.7 The PNSC shall:

a. Recommend in writing to the Plant General Manager approval or disapproval of items considered under Specification 6.5.1.6a. through d. prior to their implementation and items considered under Specification 6.5.1.6i through k.

b. Provide written notification within 24 hours to the Plant General Manager, President-Nuclear Division and the Company Nuclear Review Board of disagreement between the PNSC and the Plant General Manager; however, the Plant General Manager shall have responsibility for resolution of such disagreements pursuant to Specification 6.1.1.

DELETED

RECORDS

6.5.1.8 The PNSC shall maintain written minutes of each PNSC meeting that, at a minimum, document the results of all PNSC activities performed under the responsibility provisions of these Technical Specifications. Copies shall be provided to the President-Nuclear Division and the Company Nuclear Review Board.

6.5.2 COMPANY NUCLEAR REVIEW BOARD (CNRB)

FUNCTION

6.5.2.1 The CNRB shall function to provide independent review and audit of designated activities in the areas of:

a. Nuclear power plant operations,
b. Nuclear engineering,
c. Chemistry and radiochemistry,
d. Metallurgy,
e. Instrumentation and control,
f. Radiological safety,
g. Mechanical and electrical engineering, and
h. Quality assurance practices.

The CNRB shall report to and advise the President-Nuclear Division on those areas of responsibility specified in Specifications 6.5.2.7 and 6.5.2.8.

COMPOSITION

6.5.2.2 The President-Nuclear Division shall appoint, in writing, a minimum of five members to the CNRB and shall designate from this membership, in writing, a Chairman. The membership shall function to provide independent review and audit in the areas listed in Specification 6.5.2.1. The Chairman shall meet the requirements of ANSI/ANS - 3.1 - 1987, Section 4.7.1. The members of the CNRB shall meet the educational requirements of ANSI/ANS - 3.1 - 1987, Section 4.7.2, and have at least 5 years of professional level experience in one or more of the fields listed in Specification 6.5.2.1. CNRB members who do not possess the educational requirements of ANSI/ANS - 3.1 - 1987, Section 4.7.2 (up to a maximum of two members) shall be evaluated, and have their membership approved and documented, in writing, on a case-by-case basis by the President-Nuclear Division, considering the alternatives to the educational requirements of ANSI/ANS - 3.1 - 1987, Sections 4.1.1 and 4.1.2.

ALTERNATES

6.5.2.3 All alternate members shall be appointed in writing by the CNRB Chairman to serve on a temporary basis; however, no more than two alternates shall participate as voting members in CNRB activities at any one time.

DELETED

CONSULTANTS

6.5.2.4 Consultants shall be utilized as determined by the CNRB Chairman to provide expert advice to the CNRB.

MEETING FREQUENCY

6.5.2.5 The CNRB shall meet at least once per 6 months and as convened by the CNRB chairman or his designated alternate.

QUORUM

6.5.2.6 The quorum of the CNRB necessary for the performance of the CNRB review and audit functions of these Technical Specifications shall consist of the Chairman or his designated alternate and at least four CNRB members including alternates. No more than a minority of the quorum shall have line responsibility for operation of the facility.

REVIEW

6.5.2.7 The CNRB shall be responsible for the review of:

a. The safety evaluations for: (1) changes to procedures, equipment, or systems; and (2) tests or experiments completed under the provision of 10 CFR 50.59, to verify that such actions did not constitute an unreviewed safety question;

b. Proposed changes to procedures, equipment, or systems which involve an unreviewed safety question as defined in 10 CFR 50.59;

c. Proposed tests or experiments which involve an unreviewed safety question as defined in 10 CFR 50.59;

d. Proposed changes to Technical Specifications or this Operating License;

e. Violations of Codes, regulations, orders, Technical Specifications, license requirements, or of internal procedures or instructions having nuclear safety significance;

f. Significant operating abnormalities or deviations from normal and expected performance of unit equipment that affect nuclear safety;

g. All REPORTABLE EVENTS;

h. All recognized indications of an unanticipated deficiency in some aspect of design or operation of structures, systems, or components that could affect nuclear safety; and

i. Reports and meeting minutes of the PNSC.

AUDITS

DELETED

6.5.2.8 Audits of unit activities shall be performed under the cognizance of the CNRB. These audits shall encompass:

a. The conformance of facility operation to provisions contained within the Technical Specifications and applicable license conditions;

b. The performance, training, and qualifications of the entire facility staff;

c. The results of actions taken to correct deficiencies occurring in facility equipment, structures, systems, or method of operation that affect nuclear safety;

d. The performance of activities required by the Quality Assurance Program to meet the criteria of Appendix B, 10 CFR Part 50;

e. The fire protection programmatic controls including the implementing procedures at least once per 24 months by qualified licensee QA personnel;

f. The fire protection equipment and program implementation at least once per 12 months utilizing either a qualified offsite licensee fire protection engineer or an outside independent fire protection consultant. An outside independent fire protection consultant shall be used at least every third year;

g. The Radiological Environmental Monitoring Program and the results thereof;

h. The OFFSITE DOSE CALCULATION MANUAL and implementing procedures;

i. The PROCESS CONTROL PROGRAM and implementing procedures for processing and packaging of radioactive wastes;

j. The performance of activities required by the Quality Assurance Program for effluent and environmental monitoring;

k. The Diesel Fuel Oil Testing Program and implementing procedures; and

l. Any other area of unit operation considered appropriate by the CNRB or the President-Nuclear Division.

RECORDS

DELETED

6.5.2.9 Records of CNRB activities shall be prepared, approved, and distributed as indicated below:

a. Minutes of each CNRB meeting shall be prepared, approved, and forwarded to the President-Nuclear Division within 14 days following each meeting:

b. Reports of reviews encompassed by Specification 6.5.2.7 shall be prepared, approved, and forwarded to the President-Nuclear Division within 14 days following completion of the review: and

c. Audit reports encompassed by Specification 6.5.2.8 shall be forwarded to the President-Nuclear Division and to the management positions responsible for the areas audited within 30 days after completion of the audit by the auditing organization.

6.5.3 TECHNICAL REVIEW AND CONTROL

ACTIVITIES

6.5.3.1 Activities that affect nuclear safety shall be conducted as follows:

a. Procedures required by Specification 6.8, and other procedures that affect nuclear safety, and changes thereto, shall be prepared, reviewed, and approved. Each such procedure, or change thereto, shall be reviewed by an individual/group other than the individual/group who prepared the procedure, or change thereto, but who may be from the same organization as the individual/group who prepared the procedure, or change thereto. Procedures other than plant administrative procedures shall be approved by the Plant General Manager, Operations Manager, or the head of the department assigned responsibility for those procedures prior to implementation. The Plant General Manager shall approve plant administrative procedures and emergency plan implementing procedures. Security Plan and the implementing procedures shall be approved by Protection Services Manager prior to implementation. Changes to procedures that may involve a change to the intent of the original procedures shall be approved by the individual authorized to approve the procedure prior to implementation of the change.

ACTIVITIES (Continued) DELETED

b. Individuals responsible for reviews performed in accordance with Specification 6.5.3.1 (a) shall be members of the plant staff previously designated by the Plant General Manager and meet or exceed the minimum qualifications of ANSI N18.1-1971, Sections 4.2, 4.3.1, 4.4 and 4.6.1.

c. Each review shall include a determination of whether or not additional, cross-disciplinary review is necessary. if deemed necessary, such review shall be performed by qualified personnel of the appropriate discipline.

d. Each review will include a determination of whether or not an unreviewed safety question is involved.

6.5.3.2 Records of the above activities shall be provided to the Plant General Manager, PNSC, and/or the CNRB as necessary for required reviews.

6.6 REPORTABLE EVENT ACTION

6.6.1 The following actions shall be taken for REPORTABLE EVENTS:

a. The Commission shall be notified and a report submitted pursuant to the requirements of Section 50.73 to 10 CFR Part 50, and

b. Each REPORTABLE EVENT shall be reviewed by the PNSC, and the results of this review shall be submitted to the CNRB and the President-Nuclear Division.

6.7 SAFETY LIMIT VIOLATION

6.7.1 The following actions shall be taken in the event a Safety Limit is violated:

a. In accordance with 10 CFR 50.72, the NRC Operations Center, shall be notified by telephone as soon as practical and in all cases within one hour after the violation has been determined. The President-Nuclear Division, and the CNRB shall be notified within 24 hours.

Company Nuclear Review Board

SAFETY LIMIT VIOLATION (Continued)

b. A Licensee Event Report shall be prepared in accordance with 10 CFR 50.73.

c. The License Event Report shall be submitted to the Commission in accordance with 10 CFR 50.73, and to the CNRB, and the President-Nuclear Division within 30 days after discovery of the event.

d. Critical operation of the unit shall not be resumed until authorized by the Nuclear Regulatory Commission.

6.8 PROCEDURES AND PROGRAMS

6.8.1 Written procedures shall be established, implemented, and maintained covering the activities referenced below:

a. The applicable procedures recommended in Appendix A of Regulatory Guide 1.33, Revision 2, February 1978, Sections 5.1 and 5.3 of ANSI N18.7-1972;

b. The emergency operating procedures required to implement the requirements of NUREG-0737 and Supplement 1 to NUREG-0737 as stated in Generic Letter No. 82-33;

c. PROCESS CONTROL PROGRAM implementation;

d. OFFSITE DOSE CALCULATION MANUAL implementation;

e. Quality Control Program for effluent monitoring using the guidance in Regulatory Guide 1.21, Revision 1, June 1974;

f. Facility Fire Protection Program;

g. Quality Control Program for environmental monitoring using the guidance in Regulatory Guide 4.1, Revision 1, April 1975; and

h. Diesel Fuel Oil Testing Program implementation.

~~6.8.2 Each procedure of Specification 6.8.1 above, and changes thereto, except the Quality Control Program for environmental monitoring, shall be reviewed and approved prior to implementation and reviewed periodically as set forth in Specification 6.5.3 and administrative procedures.~~

~~6.8.3 Temporary changes to procedures of Specification 6.8.1 above may be made provided:~~

~~a. The intent of the original procedure is not altered;~~

DELETED

PROCEDURES AND PROGRAMS (Continued) DELETED

~~b. The change is approved by two members of the plant management staff, at least one of whom holds a Senior Operator license on the unit affected; and~~

~~c. The change is documented, reviewed in accordance with Specification 6.5.3 and approved by the Plant General Manager or the department head of the responsible department within 14 days of implementation.~~

6.8.4 The following programs shall be established, implemented, and maintained:

a. Primary Coolant Sources Outside Containment

A program to reduce leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. The systems include the Safety Injection System, Chemical and Volume Control System, and the Containment Spray System. The program shall include the following:

(1) Preventive maintenance and periodic visual inspection requirements, and

(2) Integrated leak test requirements for each system at refueling cycle intervals or less.

b. ~~In-Plant Radiation Monitoring~~ DELETED

~~A program which will ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions. This program shall include the following:~~

~~(1) Training of personnel,~~

~~(2) Procedures for monitoring, and~~

~~(3) Provisions for maintenance of sampling and analysis equipment.~~

c. Secondary Water Chemistry

A program for monitoring of secondary water chemistry to inhibit steam generator tube degradation. This program shall include:

(1) Identification of a sampling schedule for the critical variables and control points for these variables,

(2) Identification of the procedures used to measure the values of the critical variables,

PROCEDURES AND PROGRAMS (Continued)

(3) Identification of process sampling points, which shall include monitoring the discharge of the condensate pumps for evidence of condenser in-leakage,

(4) Procedures for the recording and management of data,

(5) Procedures defining corrective actions for all off-control point chemistry conditions, and

(6) A procedure identifying: (a) the authority responsible for the interpretation of the data, and (b) the sequence and timing of administrative events required to initiate corrective action.

d. Post-Accident Sampling

A program which will ensure the capability to obtain and analyze reactor coolant, radioactive iodines and particulates in plant gaseous effluents, and containment atmosphere samples under accident conditions. The program shall include the following:

(1) Training of personnel,

(2) Procedures for sampling and analysis, and

(3) Provisions for maintenance of sampling and analysis equipment.

e. Diesel Fuel Oil Testing Program

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

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

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

   2. a flash point and kinematic viscosity within limits for Grade No. 2-D fuel oil per ASTM D975, and

   3. a clear and bright appearance with proper color;

b. Other properties for Grade No. 2-D fuel oil per ASTM D975 are within limits within 30 days following sampling and addition to storage tanks; and

c. Total particulate concentration of the fuel oil is ≤ 10 mg/liter when tested every 31 days in accordance with either ASTM D-2276 or ASTM D-5452.

f. Radioactive Effluent Controls Program

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

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

2. Limitations on the concentrations of radioactive material released in liquid effluents to UNRESTRICTED AREAS, conforming to ten times the 10 CFR 20, Appendix B, Table 2, Column 2 limits;

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

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

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

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

7. Limitations on the dose rate resulting from radioactive material released in gaseous effluents to areas beyond the site boundary to 500 mrem per year to the whole body, 3000 mrem per year to the skin and 1500 mrem per year to any organ from iodine 131, iodine 133 tritium and all radionuclides in particulate form with half live greater than 8 days.

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

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

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

g. Radiological Environmental Monitoring Program

A program shall be provided to monitor the radiation and radionuclides in the environs of the plant. The program shall provide (1) representative measurements of radioactivity in the highest potential exposure pathways, and (2) verification of the accuracy of the effluent monitoring program and modeling of environmental exposure pathways. The program shall (1) be contained in the ODCM, (2) conform to the guidance of Appendix I to 10 CFR Part 50, and (3) include the following:

DELETED

1. Monitoring, sampling, analysis, and reporting of radiation and radionuclides in the environment in accordance with the methodology and parameters in the ODCM,

2. A Land Use Census to ensure that changes in the use of areas at and beyond the SITE BOUNDARY are identified and that modifications to the monitoring program are made if required by the results of this census, and

3. Participation in a Interlaboratory Comparison Program to ensure that independent checks on the precision and accuracy of the measurements of radioactive materials in environmental sample matrices are performed as part of the quality assurance program for environmental monitoring.

h. Containment Leakage Rate Testing Program

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

1) Type A tests will be performed either in accordance with Bechtel Topical Report BN-TOP-1, Revision 1, dated November 1, 1972, or the guidelines of Regulatory Guide 1.163.

2) A vacuum test will be performed in lieu of a pressure test for airlock door seals at the required intervals (Amendment Nos. 73 and 77, issued by NRC November 11, 1981).

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 49.9 psig.

The maximum allowable containment leakage rate, L_a, at P_a, shall be 0.25% of containment air weight per day.

Leakage Rate acceptance criteria are:

1) The As-found containment leakage rate acceptance criterion is $\leq 1.0\ L_a$. Prior to increasing primary coolant temperature above 200°F following testing in accordance with this program or restoration from exceeding $1.0\ L_a$, the As-left leakage rate acceptance criterion is $\leq 0.75\ L_a$, for Type A test.

2) The combined leakage rate for all penetrations subject to Type B or Type C testing is as follows:

 - The combined As-left leakage rates determined on a maximum pathway leakage rate basis for all penetrations shall be verified to be less than $0.60\ L_a$, prior to increasing primary coolant temperature above 200°F following an outage or shutdown that included Type B and Type C testing only.

 - The As-found leakage rates, determined on a minimum pathway leakage rate basis, for all newly tested penetrations when summed with the As-left minimum pathway leakage rate leakage rates for all other penetrations shall be less than $0.6\ L_a$, at all times when containment integrity is required.

3) Overall air lock leakage acceptance criteria is $\leq 0.05\ L_a$, when pressurized to P_a.

The provisions of Specification 4.0.2 do not apply to the test frequencies contained within the Containment Leakage Rate Testing Program.

6.8.5 — INSERT

6.9 REPORTING REQUIREMENTS

ROUTINE REPORTS

6.9.1 In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the U.S. Nuclear Regulatory Commission, Document Control Desk, Washington, DC pursuant to 10 CFR 50.4.

STARTUP REPORT

6.9.1.1 A summary report of plant startup and power escalation testing shall be submitted following: (1) receipt of an Operating License, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the unit.

INSERT

"6.8.5 Administrative procedures shall be developed and implemented to limit the working hours of plant staff who perform safety-related functions e.g. licensed Senior Operators, licensed Operators, health physicists, auxiliary operators, and key maintenance personnel. The procedures shall include guidelines on working hours that ensure that adequate shift coverage is maintained without routine heavy use of overtime for individuals.

Any deviation from the working hour guidelines shall be authorized by the applicable department manager or higher levels of management, in accordance with established procedures and with documentation of the basis for granting the deviation. Controls shall be included in the procedures such that individual overtime shall be reviewed monthly by the Plant General Manager or his designee to assure that excessive hours have not been assigned. Routine deviation from the working hour guidelines shall not be authorized."

STARTUP REPORT (Continued)

The report shall address each of the tests identified in the FSAR and shall in general include a description of the measured values of the operating conditions of characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report. Subsequent Startup Reports shall address startup tests that are necessary to demonstrate the acceptability of changes and/or modifications.

Startup Reports shall be submitted within: (1) 90 days following completion of the Startup Test Program, (2) 90 days following resumption or commencement of commercial power operation, or (3) 9 months following initial criticality, whichever is earliest. If the Startup Report does not cover all three events (i.e., initial criticality, completion of Startup Test Program, and resumption or commencement of commercial operation), supplementary reports shall be submitted at least every 3 months until all three events have been completed.

ANNUAL REPORTS*

6.9.1.2 Annual Reports covering the activities of the unit as described below for the previous calendar year shall be submitted prior to March 1 of each year.

Reports required on an annual basis shall include:

a. A tabulation on an annual basis of the number of station, utility, and other personnel (including contractors) for whom monitoring was required, receiving annual deep dose equivalent exposures greater than 100 mrem/yr and their associated man-rem exposure according to work and job functions** (e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling). The dose assignments to various duty functions may be estimated based on pocket dosimeter, thermoluminescent dosimeter (TLD), or film badge measurements. Small exposures totalling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total deep dose equivalent received from external sources should be assigned to specific major work functions;

b. The results of specific activity analyses in which the primary coolant exceeded the limits of Specification 3.4.8. The following information shall be included: (1) Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded (in graphic and tabular format); (2) Fuel burnup by core region; (3) Clean-up flow history starting 48 hours prior to the first sample in which the limit was exceeded; (4) History of degassing operations, if any, starting 48 hours prior to the first sample in which the limit was exceeded; and (5) The time duration when the specific activity of the primary coolant exceeded 1.0 microcurie per gram DOSE EQUIVALENT I-131.

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

**This tabulation supplements the requirements of § 20.2206 of 10 CFR Part 20.

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT*

6.9.1.3 The Annual Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted by May 15 of each year. The report shall include summaries, interpretations, and analyses of trends of the results of the radiological environmental monitoring program for the reporting period. The material provided shall be consistent with the objectives outlined in (1) the Offsite Dose Calculation Manual (ODCM), and in (2) 10 CFR 50, Appendix I, Sections IV.B.2, IV.B.3, and IV.C.

The Annual Radiological Environmental Operating Report shall include the results of analyses of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the table and figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted in a supplementary report as soon as possible.

6.9.1.4 RADIOACTIVE EFFLUENT RELEASE REPORT**

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

MONTHLY OPERATING REPORTS

6.9.1.5 Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the PORVs or safety valves, shall be submitted on a monthly basis to the U.S. Nuclear Regulatory Commission, Document Control Desk, Washington, D.C. 20555, with a copy to the Regional Administrator of the Regional Office of the NRC, no later than the 15th of each month following the calendar month covered by the report.

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*A single submittal may be made for a multiple unit station.

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

PEAKING FACTOR LIMIT REPORT

6.9.1.6 The W(Z) function(s) for Base-Load Operation corresponding to a ±2% band about the target flux difference and/or a ±3% band about the target flux difference, the Load-Follow function $F_Z(Z)$ and the augmented surveillance turnon power fraction, P_T, shall be provided to the U.S. Nuclear Regulatory Commission, whenever P_T is <1.0. In the event, the option of Baseload Operation (as defined in Section 4.2.2.3) will not be exercised, the submission of the W(Z) function is not required. Should these values (i.e., W(Z), $F_Z(Z)$ and P_T) change requiring a new submittal or an amended submittal to the Peaking Factor Limit Report, the Peaking Factor Limit Report shall be provided to the NRC Document Control desk with copies to the Regional Administrator and the Resident Inspector within 30 days of their implementation, unless otherwise approved by the Commission.

The analytical methods used to generate the Peaking Factor limits shall be those previously reviewed and approved by the NRC. If changes to these methods are deemed necessary they will be evaluated in accordance with 10 CFR 50.59 and submitted to the NRC for review and approval prior to their use if the change is determined to involve an unreviewed safety question or if such a change would require amendment of previously submitted documentation.

CORE OPERATING LIMITS REPORT

6.9.1.7 Core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT (COLR) before each reload cycle or any remaining part of a reload cycle for the following:

1. Axial Flux Difference for Specification 3.2.1.
2. Control Rod Insertion Limits for Specification 3.1.3.6.
3. Heat Flux Hot Channel Factor - $F_Q(Z)$ for Specification 3/4.2.2.
4. All Rods Out position for Specification 3.1.3.2.
5. Nuclear Enthalpy Rise Hot Channel Factor for Specification 3/4.2.3

The analytical methods used to determine the AFD limits shall be those previously reviewed and approved by the NRC in:

1. WCAP-10216-P-A, "RELAXATION OF CONSTANT AXIAL OFFSET CONTROL F_Q SURVEILLANCE TECHNICAL SPECIFICATION," June 1983.

2. WCAP-8385, "POWER DISTRIBUTION CONTROL AND LOAD FOLLOWING PROCEDURES - TOPICAL REPORT," September 1974.

The analytical methods used to determine $F_Q(Z)$, $F_{\Delta}H$ and the K(Z) curve shall be those previously reviewed and approved by the NRC in:

1. WCAP-9220-P-A, Rev. 1, "Westinghouse ECCS Evaluation Model - 1981 Version," February 1982.

2. WCAP-10054-P-A, (proprietary), "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code", August 1985.

3. WCAP-10054-P, Addendum 2, Revision 1 (proprietary), "Addendum to the Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code: Safety Injection in the Broken Loop and Improved Condensation Model", October 1995.*

4. WCAP-12945-P, "Westinghouse Code Qualification Document For Best Estimate LOCA Analysis," Volumes I-V, June 1996.**

5. USNRC Safety Evaluation Report, Letter from R. C. Jones (USNRC) to N. J. Liparulo (W), "Acceptance for Referencing of the Topical Report WCAP-12945(P) 'Westinghouse Code Qualification Document for Best Estimate Loss of Coolant Analysis,'" June 28, 1996.**

6. Letter dated June 13, 1996, from N. J. Liparulo (W) to Frank R. Orr (USNRC), "Re-Analysis Work Plans Using Final Best Estimate Methodology."**

7. WCAP-12610-P-A, "VANTAGE+ Fuel Assembly Reference Core Report," S. L. Davidson and T. L. Ryan, April 1995.

The analytical methods used to determine Rod Bank Insertion Limits and the All Rods Out position shall be those previously reviewed and approved by the NRC in:

1. WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985.

The ability to calculate the COLR nuclear design parameters are demonstrated in:

1. Florida Power & Light Company Topical Report NF-TR-95-01, "Nuclear Physics Methodology for Reload Design of Turkey Point & St. Lucie Nuclear Plants".

Topical Report NF-TR-95-01 was approved by the NRC for use by Florida Power & Light Company in:

1. Safety Evaluation by the Office of Nuclear Reactor Regulations Related to Amendment No. 174 to Facility Operating License DPR-31 and Amendment No. 168 to Facility Operating License DPR-41, Florida Power & Light Company Turkey Point Units 3 and 4, Docket Nos. 50-250 and 50-251.

The AFD, $F_Q(Z)$, $F_{\Delta}H$, K(Z), and Rod Bank Insertion Limits shall be determined such that all applicable limits of the safety analyses are met. The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector, unless otherwise approved by the Commission.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Regional Administrator of the Regional Office of the NRC within the time period specified for each report as stated in the Specifications within Sections 3.0, 4.0, or 5.0.

*This reference is only to be used subsequent to NRC approval.
**As evaluated in NRC Safety Evaluation dated December 20, 1997.

DELETED

6.10 RECORD RETENTION

6.10.1 In addition to the applicable record retention requirements of Title 10, Code of Federal Regulations, the following records shall be retained for at least the minimum period indicated.

6.10.2 The following records shall be retained for at least 5 years:

a. Records and logs of unit operation covering time interval at each power level;

b. Records and logs of principal maintenance activities, inspections, repair, and replacement of principal items of equipment related to nuclear safety;

c. All REPORTABLE EVENTS;

d. Records of surveillance activities, inspections, and calibrations required by these Technical Specifications;

e. Records of changes made to the procedures required by Specification 6.8.1;

f. Records of radioactive shipments;

g. Records of sealed source and fission detector leak tests and results; and

h. Records of annual physical inventory of all sealed source material of record.

6.10.3 The following records shall be retained for the duration of the unit Operating License:

a. Records and drawing changes reflecting unit design modifications made to systems and equipment described in the Final Safety Analysis Report;

b. Records of new and irradiated fuel inventory, fuel transfers, and assembly burnup histories;

c. Records of facility radiation and contamination surveys;

d. Records of radiation exposure for all individuals entering radiation control areas;

e. Records of gaseous and liquid radioactive material released to the environs;

f. Records of transient or operational cycles for those unit components identified in Table 5.7-1;

g. Records of reactor tests and experiments;

h. Records of training and qualification for current members of the facility staff;

~~RECORD RETENTION (Continued)~~ DELETED

i. ~~Records of inservice inspections performed pursuant to these Technical Specifications;~~

j. ~~Records of quality assurance activities required for the duration of the unit Operating License by the Quality Assurance Manual;~~

k. ~~Records of reviews performed for changes made to procedures or equipment or reviews of tests and experiments pursuant to 10 CFR 50.59;~~

l. ~~Records of meetings of the PNSC and the CNRB;~~

m. ~~Records of the service lives of all hydraulic and mechanical snubbers required by Specification 3.7.6 including the date at which the service life commences and associated installation and maintenance records;~~

n. ~~Records of secondary water sampling and water quality; and~~

o. ~~Annual Radiological Environmental Operating Reports and records of analyses transmitted to the licensee which are used to prepare the Annual Radiological Environmental Operating Report.~~

p. ~~Records for Environmental Qualification which are covered under the provisions of 10 CFR 50.49.~~

q. ~~Records of reviews performed for changes made to the Offsite Dose Calculation Manual and the Process Control Program.~~

~~6.11 RADIATION PROTECTION PROGRAM~~

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~~6.11.1 Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained, and adhered to for all operations involving personnel radiation exposure.~~

6.12 HIGH RADIATION AREA

6.12.1 Pursuant to paragraph 20.1601(c) of 10 CFR Part 20, in lieu of the "control device" or "alarm signal" required by paragraph 20.1601(a), each high radiation area, as defined in 10 CFR Part 20, in which the intensity of radiation is greater than 100 mrem/hr but equal to or less than 1000 mrem/hr at 30 cm (12 in.) from the radiation source or from any surface which the radiation penetrates shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit (RWP). Individuals qualified in radiation protection procedures (e.g., Health Physics Technician) or personnel continuously escorted by such individuals may be exempt from the RWP issuance requirement during the performance of their assigned duties in high radiation areas with exposure rates equal to or less than 1000 mrem/hr, provided they are otherwise following plant radiation protection procedures for entry into such high radiation areas. ¶ Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

New paragraph

HIGH RADIATION AREA (Continued)

a. A radiation monitoring device which continuously indicates the radiation dose rate in the area; or

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

c. An individual qualified in radiation protection procedures with a radiation dose rate monitoring device, who is responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by the Health Physics Shift Supervisor in the RWP.

6.12.2 In addition to the requirements of Specification 6.12.1, areas accessible to personnel with radiation levels greater than 1000 mrem/hr at 30 cm (12 in.) and less than 500 rads/hr at 1 meter from the radiation source or from any surface which the radiation penetrates shall be provided with locked doors to prevent unauthorized entry, and the keys shall be maintained under the administrative control of the shift supervisor on duty and/or health physics supervision. Doors shall remain locked except during periods of access by personnel under an approved RWP which shall specify the dose rate levels in the immediate work areas and the maximum allowable stay time for individuals in that area. In lieu of the stay time specification of the RWP, direct or remote (such as closed circuit TV cameras) continuous surveillance may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities being performed within the area.

For individual high radiation areas accessible to personnel with radiation levels of greater than 1000 mrem/hr and less than 500 rads/hr that are located within large areas, such as PWR containment, where no enclosure exists for purposes of locking, and where no enclosure can be reasonably constructed around the individual area, that individual area shall be barricaded, conspicuously posted, and a flashing light shall be activated as a warning device.

~~6.13 PROCESS CONTROL PROGRAM (PCP)~~

~~6.13.1 The Process Control Program (PCP) shall contain the current formulas, sampling, analyses, test, and determinations to be made to ensure that processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10 CFR Parts 20, 61, and 71, State regulations, burial ground requirements, and other requirements governing the disposal of solid radioactive waste.~~

DELETED

PROCESS CONTROL PROGRAM (Continued)

DELETED

6.13.2 Licensee-initiated changes to the PCP:

a. Shall be documented and records of reviews performed shall be retained as required by Specification 6.10.3q. This documentation shall contain:

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

2) A determination that the change will maintain the overall conformance of the solidified waste product to existing requirements of Federal, State, or other applicable regulations.

b. Shall become effective after review and acceptance by the PNSC and the approval of the Plant General Manager.

6.14 OFFSITE DOSE CALCULATION MANUAL (ODCM)

6.14.1 The ODCM shall contain the following:

a. The methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm and trip setpoints, and in the conduct of the Radiological Environmental Monitoring Program; and

b. The radioactive effluent controls and radiological environmental monitoring activities, and descriptions of the information that should be included in the Annual Radiological Environmental Operating, and Annual Radioactive Effluent Release Reports required by Specification 6.9.1.3 and Specification 6.9.1.4.

6.14.2 Licensee initiated changes to the ODCM:

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a. Shall be documented and records of reviews performed shall be retained ~~as required by specification 6.10.3q~~. This documentation shall contain:

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

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

DELETED

b. Shall become effective after ~~the review and acceptance by the PNSC and~~ approval of the Plant General Manager; and

OFFSITE DOSE CALCULATION MANUAL (ODCM) (Continued)

c. Shall be submitted to the NRC in the form of a complete, legible copy of the entire ODCM as a part of or concurrent with the Annual Radioactive Effluent Release Report for the period of the report in which any change in the ODCM was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed, and shall indicate the date (i.e., month and year) the change was implemented.

ATTACHMENT 1

DESCRIPTION OF AMENDMENTS REQUEST

1.0 Purpose

Florida Power and Light Company (FPL) requests that Appendix A of Facility Operating License DPR-31 and DPR-41 be amended to modify Technical Specifications (TS) Table 3.9-1, "Spent Fuel Burnup Requirements for Storage in Region II of the Spent Fuel Pit," and 5.6.1, "Criticality." The proposed changes seek to take advantage of improvements in the NRC approved Westinghouse methodology which credits soluble boron and provides a direct method for ensuring spent fuel rack subcriticality.

The proposed changes will enhance FPL's ability to retain the use of spent fuel storage cells in the Spent Fuel Pool (SFP) in the event of further Boraflex degradation, thereby preserving the ability to store spent fuel in the SFPs without the need for out of pool storage capability.

2.0 Background

The industry has experienced issues related to the degradation of Boraflex, e.g. dissolution of B^{10}, gap formation, and shrinkage. NRC had requested the licensees via Generic Letter (GL) 96-04 "Boraflex Degradation in Spent Fuel Pool Storage Racks" to (1) assess the capability of the Boraflex to maintain a 5 % subcriticality margin and (2) submit to the NRC a plan describing its proposed actions if this subcriticality margin cannot be maintained by Boraflex material because of current or projected future Boraflex degradation. In response to the request, Turkey Point had assessed Boraflex degradation by considering the integrated gamma dose to the Boraflex panels in conjunction with the Blackness Testing results and the monitoring of silica level in the SFP.

The response to the GL concluded that at Turkey Point, no further Boraflex degradation could occur due to gamma irradiation. It is known that gamma irradiation from the spent fuel results in the formation of gaps and the shrinkage of the Boraflex panels. The saturation gamma dose level for the Boraflex degradation mechanism has been reported by EPRI to be between $9x10^9$ and $1.5x10^{10}$ rads. The calculated gamma dose to irradiated Boraflex panels in Turkey Point Units 3 and 4 SFPs is beyond the saturation level for this Boraflex degradation mechanism. Therefore, no additional significant shrinkage is expected due to gamma exposure beyond the saturation level.

The GL response also outlined the long-term remedies if a subcriticality margin of 5 % could not be maintained because of Boraflex degradation. FPL had indicated that should Boraflex degradation impact the capability to maintain $K_{eff} \leq 0.95$, FPL would consider taking credit for soluble boron in the SFP. Furthermore, it was stated in the response that FPL would evaluate the Westinghouse soluble boron credit methodology for reanalyzing the spent fuel rack criticality for Turkey Point Units 3 & 4 for crediting soluble boron.

3.0 Need for the Proposed Technical Specification Change

FPL has an on-going in-service Boraflex verification program. The goals of the program are to confirm the in-service Boraflex panel performance data in terms of gap formation, gap distribution, and gap size. The program accomplishes these goals through the performance of the Blackness Testing, and the tracking of the SFP silica levels.

Blackness Testing is performed on a test frequency of every five years on specific Boraflex panels in either SFP. Engineering evaluations are performed to evaluate the results of the Blackness Testing to demonstrate that the subcritical margin required by Technical Specifications for the storage of fuel in the Spent Fuel Pool continues to be met. The results of these tests have demonstrated that the required TS subcritical margin for the storage of fuel in the SFP has been met.

The proposed changes to Turkey Point Units 3 and 4 TS Table 3.9-1 and Section 5.6.1, seek to take advantage of the NRC approved Westinghouse methodology which provides a direct method for ensuring subcriticality by utilizing the soluble boron that is contained in the SFP.

TS Table 3.9-1 will be revised to provide less restrictive enrichment and burnup requirements for the storage of fuel assemblies and TS 5.6.1 will be revised to provide boron concentration requirements for the water contained in the SFP. Westinghouse has performed criticality analyses for the Spent Fuel Storage for Turkey Point Units 3 & 4 and submitted here as Attachments 4 and 5. The Turkey Point Units 3 & 4 Boron Dilution analysis is submitted here as Attachment 6.

The criticality analyses presented in Attachment 7 change the analytical basis for the Fresh Fuel storage without the need to change the TS 5.6.1.2.

4.0 System Description and Design Basis Requirements

The design basis of the SFP is to provide for the safe storage of irradiated fuel assemblies. The pool is filled with borated water. The water removes decay heat, provides shielding for the personnel handling the fuel, and reduces the amount of radioactive gases released during a fuel handling accident. The water volume corresponding to the TS minimum level elevation of 56′-10″ is conservatively determined to be 264,000 gallons.

Turkey Point Units 3 and 4 SFPs are each divided in two regions. Region I is designed to store high reactivity fuel (fresh or low burnup fuel). Region II can store lower reactivity (lower enrichment or higher burnup), but at a higher storage density. The maximum number of fuel cell locations is 1404. The spent fuel racks are designed to support and protect the spent fuel assemblies under normal and credible accident conditions. The SFP boron concentration is typically in the range of 2000 to 2100 ppm. Existing TS 3.9.14 states that the minimum boron concentration in the Spent Fuel Pit shall be 1950 ppm at all times when fuel is stored in the Spent Fuel Pit.

Criticality of fuel assemblies in a fuel storage rack is prevented by the design of the rack, which limits fuel assembly interaction. The SFP is maintained in a subcritical configuration in unborated water by:

- Limiting the fuel enrichment to no greater than 4.5 weight percent (w/o) U^{235},
- Maintaining a fixed separation between individual fuel assemblies stored in cells arranged into two separate regions (Regions I and II),
- Interposition of neutron absorbing panels (Boraflex) between assemblies, and
- Specifying assignment of fuel assemblies to a physical location based on accumulated burnup criteria.

5.0 Technical Specification Change Request

The proposed changes are described below and a markup of the proposed changes is provided in Attachment 3:

1. TS 3/4.9.14, "Table 3.9-1 Spent Fuel Burnup Requirements for Storage in Region II of the Spent Fuel Pit": Replace existing TS Table 3.9-1 with the revised Table 3.9-1 which provides the burnup for a given enrichment up to 4.5 w/o of U^{235} that is allowed to be stored in Region II.

   Justification: The Turkey Point spent fuel racks have been analyzed to take credit for soluble boron including the presence of Boraflex poison panels in the spent fuel rack. The maximum enrichment limit is increased from 1.5 w/o to 1.6 w/o of U^{235}, and the required burnups for storage of initial enrichments up to 4.5 w/o of U^{235} are updated. Storage of fuel assemblies with initial enrichments higher than 1.6 w/o of U^{235} in all cells of the Turkey Point Units 3 and 4 Region II fuel racks is achievable by means of burnup credit using reactivity equivalencing. The concept of equivalencing is based on reactivity decrease associated with fuel depletion. For burnup credit a series of reactivity calculations has been performed assuming conservative conditions for the fuel storage cells (Attachment 4) to generate a set of enrichment-fuel assembly discharge burnup ordered pairs which all yield an equivalent K_{eff} when stored in the spent fuel storage racks in Region II. The limit of 1950 ppm, bounds the value of 650 ppm for normal conditions, which is required to maintain the spent fuel storage rack $K_{eff} \leq 0.95$ when fuel assemblies are stored in accordance with TS 3/4.9.14.TS Table 3.9-1 represents combinations of fuel enrichment and discharge burnup based on calculations of constant rack reactivity with a soluble boron requirement of at least 650 ppm.

2. TS 5.6.1.1 "Criticality": Renumber 5.6.1.1a to 5.6.1.1b, 5.6.1.1b to 5.6.1.1c, and 5.6.1.1c to 5.6.1.1d.

   Justification: This is an administrative change. Renumbering of these sections is needed in order to incorporate a new TS section numbered as 5.6.1.1a.

3. TS 5.6.1.1a: Add 5.6.1.1a to read, "A K_{eff} equivalent to less than 1.0 when flooded with unborated water, which includes a conservative allowance for uncertainties."

Justification: TS 5.6.1.1a is added in accordance with the "Westinghouse Spent Fuel Rack Criticality Analysis Methodology" SFP criteria without the presence of soluble boron. For both Region I and Region II, K_{eff} is required to remain less than 1.0 without soluble boron present in the SFP water ensuring that the spent fuel racks will remain subcritical under conditions when all cells are loaded with fuel assemblies with permitted enrichments and fuel burnups. The analytical criterion of K_{eff} equivalent to less than 1.0 when flooded with unborated water has been specifically approved by the NRC for this application.

4. TS 5.6.1.1b: Revise 5.6.1.1b to read, "A K_{eff} equivalent to less than or equal to 0.95 when flooded with water borated to 650 ppm, which includes a conservative allowance for uncertainties."

   Justification: Delete the current reactivity limitations of flooded conditions with unborated water. In addition, the conservative allowances of 0.97% $\Delta k/k$ for Region I and 1.96% $\Delta k/k$ for Region II are addressed in the Turkey Point specific analyses using the approved Westinghouse methodology and are not included in the TS. These limitations are replaced with K_{eff} limits that are based on the SFP criterion for the minimum soluble boron requirements to maintain subcriticality in accordance with the "Westinghouse Spent Fuel Rack Criticality Analysis Methodology" described in WCAP-14416-NP-A, Rev 1. The minimum soluble boron of 650 ppm is the required soluble boron limit which has been calculated by Westinghouse specifically for Turkey Point Units 3 and 4 and the supporting analyses are presented in Attachment 4. The TS limit of 1950 ppm, bounds the value of 650 ppm for normal conditions, which is required to maintain the spent fuel storage rack $K_{eff} \leq 0.95$ when fuel assemblies are stored in accordance with TS 3/4.9.14. The analysis uncertainties will be incorporated into the UFSAR consistent with NUREG 1431.

5. TS 5.6.1.1c: Change the Region "1" to read " I" and Region 2 to read "II".

   Justification: This is an administrative change to have consistent naming of SFP Regions I and II.

6. TS 5.6.1.2: There is no change required for the TS 5.6.1.2.

   Justification: The criticality analyses for Fresh Fuel storage racks for Turkey Point Units 3 and 4 constitute an update to the safety analyses bases. The safety analyses supporting the current TS for Fresh Fuel Storage racks has been updated as presented in Attachment 7 and in accordance to the Westinghouse methodology described in WCAP 14416-NP-A, Revision 1. The results of the analyses in Attachment 7 meet the current TS acceptance criteria for the effective neutron multiplication factor including uncertainties for fully flooded and for optimum moderation conditions. Therefore, there is no need to change the current TS.

6.0 Safety Analysis

6.1 Westinghouse Spent Fuel Rack Criticality Analysis Methodology

The methodology which allows credit for soluble boron is contained in the topical report "Westinghouse Spent Fuel Rack Criticality Analysis Methodology" described in WCAP-14416-NP-A, Revision 1. This methodology was approved by a NRC Safety Evaluation dated October 25, 1996.

Criticality of fuel assemblies in a fuel storage rack is prevented by the design of the rack, which limits fuel assembly interaction. This is done by fixing the minimum separation between fuel assemblies and placing absorber panels between storage cells. The design basis for preventing criticality in the SFP or for fresh fuel storage is examined on a 95/95 basis. The 95/95 basis is defined as the upper limit, with a 95 percent probability at a 95 percent confidence level, of effective neutron multiplication factor K_{eff} of the fuel assembly array, including uncertainties and manufacturing tolerances.

For the SFP, the criteria with the credit for soluble boron are:

1. The 95/95 basis K_{eff} for the storage will be less than 1.0 without the presence of the soluble boron.
2. The 95/95 basis K_{eff} for the storage will be less than or equal to 0.95 with the presence of a defined level of soluble boron in the SFP.

These criteria are defined in the topical report in WCAP-14416-NP-A, Revision 1. The analyses require the reactivity of the racks to be below 1.0 with all manufacturing tolerances and uncertainties and without any credit for soluble boron. Credit for soluble boron is taken to provide safety margin by maintaining rack reactivity to be approximately 5% below that for the no boron case, but also less than or equal to 0.95, including uncertainties and tolerances. Credit for soluble boron is also taken under accident scenarios.

6.2 Turkey Point Specific Calculations

The methodology employed in Turkey Point spent fuel rack criticality calculations is based on the NRC approved "Westinghouse Spent Fuel Rack Criticality Analysis Methodology" described in WCAP-14416-NP-A, Revision 1.

The Turkey Point Units 3 & 4 spent fuel racks have been analyzed to allow storage of Westinghouse 15x15 fuel assemblies in all cells in Regions I and in all cells in Region II of the SFP. The K_{eff} was calculated for Region I and Region II. Westinghouse analyzed two cases for each region, one case with no soluble boron, and one case with credit for soluble boron. These analyses include the presence of Boraflex poison panels in the spent fuel rack. However, the Boraflex panels are degraded by shrinkage, which creates gaps in the panels. The storage racks were analyzed assuming Boraflex degradation to account for gap formation, shrinkage, and loss of B^{10} from the Boraflex panel via dissolution.

A Boron Dilution analysis was performed by Westinghouse for crediting soluble boron in the Turkey Point Units 3 & 4 spent fuel rack criticality analysis. The boron dilution analysis was performed to ensure that sufficient time is available to detect and mitigate a dilution event before the spent fuel rack criticality analysis 0.95 K_{eff} design basis is exceeded.

Attachment 4 contains the Turkey Point specific calculations "Criticality for Spent Fuel Storage for Turkey Point Units 3 & 4 (Degraded Boraflex)." Attachment 5 contains the criticality analyses results for the more limiting cases (no soluble boron for Region I and Region II) which assume loss of B^{10} from the Boraflex panels via dissolution. Attachment 6 contains the SFP Boron Dilution analysis for Turkey Point.

6.3 Analytical Assumptions

The design method, which insures the criticality safety of fuel assemblies in the fuel storage rack, is described in detail in the Westinghouse topical report WCAP-14416-NP-A, Revision 1. In Attachment 4, Westinghouse describes the computer codes, the benchmarking, and the methodology which are used to calculate the criticality safety limits for Turkey Point Units 3 & 4. The specific assumptions employed in the Turkey Point criticality analyses are listed in the related sections of Attachments 4 and 5. The assumptions made for the Boron Dilution calculations are listed in Attachment 6.

In addition to crediting soluble boron in the SFP criticality analysis, the following storage configurations and enrichment limits were evaluated in the spent fuel rack criticality analyses:

- Westinghouse 15X15 fuel assemblies with nominal enrichments less than or equal to 4.50 w/o U^{235} with 6 inch long natural uranium axial blankets at the top and bottom of the fuel rods, can be stored in any cell location in Region I.

- Westinghouse 15X15 fuel assemblies with initial nominal enrichments less than or equal to 1.60 w/o U^{235} can be stored in any cell location in Region II.

- Westinghouse 15X15 fuel assemblies with initial nominal enrichments greater than 1.60 w/o U^{235} must satisfy a minimum burnup requirement as shown in the revised TS Table 3.9-1.

- The characteristics of the Westinghouse 15x15 Optimized Fuel Assembly (OFA) Debris Resistant Fuel Assembly (DRFA) fuel assemblies were assumed in the spent fuel rack criticality analysis, since they bound storage of the Westinghouse 15X15 OFA and Low Parasitic Absorber Rod (LOPAR) fuel assemblies.

6.4 Validation of the Analytical Assumptions

Boraflex degradation is monitored by the Blackness Testing and the level of silica in the water of the SFP. The measured data is used to validate the assumptions used in the criticality analyses and to ensure that they conservatively bound the actual Boraflex conditions in the Turkey Point SFP.

The input and assumptions on uncertainties and tolerances for the storage racks criticality analyses and the input used in the boron dulution analyses have been documented and independently verified in accordance to FPL's Quality Assurance program.

6.4.1 Blackness Testing Results

The Blackness Testing results for each unit and both of their respective storage regions have been consolidated for the following reasons:

- The radiation and chemical environment that cause the formation of gaps and shrinkage are similar,
- The Boraflex material in both storage regions is made from the same polymer.

Blackness Testing has been performed for a fraction of the total Boraflex panels in the SFP. However, the measured results conservatively represent the conditions of irradiated Boraflex panels in the SFP for the following reasons:

- The manufacturing process for the spent fuel storage racks was very similar,
- The thermal and chemical processes for every storage cell are very similar,
- The degradation mechanism by gamma irradiation is the same for every cell, and
- The measured cells were chosen to provide information representative of the dose at which the irradiation effects plateau.

The consolidated Blackness testing results showed the following characteristics about Boraflex gaps and shrinkage:

- Less than 60% of all panels measured had gaps.
- The number of gaps per panel ranged from 0 to 7.
- At least 95% of the panels tested showed that there were 3 gaps or less.
- The largest gap width was 2.4 inches and the smallest gap was 0.5 inches.
- At least 95% of the gap widths were 1.5 inches or less.
- The axial location of these gaps along the panel was randomly distributed and the median distance between gaps was 44 inches.
- The nominal panel length had decreased by 2 inches.

A conservative representation of the entire SFP is taken in the analysis by assuming all cells having the following characteristics for each Boraflex panel:

- Five gaps/panel are assumed, with the width of each gap being 1.5 inches, separated by 7.5 inches (center to center). This assumption is conservative because it assumes 5 gaps in every panel whereas 40% of the panels tested showed no gaps.
- The length of each Boraflex panel is assumed to shrink by approximately 4.2 inches, which is greater than the measured shrinkage of 2.0 inches.
- The measured results yield a value of 2.1% of Boraflex unavailable for neutron absorption compared to the analysis value of 8.2 %.
- The gaps in the model are at the same axial position in each panel, which significantly increases the neutronic communication between cells. This does not occur in the actual SFP since the gaps are randomly distributed along the axial length of the Boraflex panel.

6.4.2 Silica Monitoring

Silica is a component of Boraflex. Dissolution of silica from the Boraflex into the main body of the water of the Spent Fuel Pool will occur until it reaches an equilibrium concentration. One way to disturb this equilibrium condition and encourage further degradation is via the Spent Fuel Purification System that removes impurities from the water via a mixed bed ion exchanger. There are three factors limiting dissolution of silica from the Boraflex:

- The free flow of Spent Fuel Pool water inside the Boraflex wrapper plate is very limited due to the construction of the Boraflex wrapper plate and its attachment to the wall of the fuel storage cell,
- There are very few exchange sites available in the ion exchange resin for the removal of silica causing the resin to saturate with silica quickly which results in only a slight decrease in silica concentration, and
- The purification system is normally lined up to the Refueling Water Storage Tank.

The most significant method of decreasing silica concentration in the SFP is via a feed and bleed operation, which is an evolution that is rarely used.

Dissolution of B^{10} out of the Boraflex as a degradation mechanism is not considered a significant problem. This is evidenced by the low concentration of silica in the SFP water as a function of time. Attachment 8 provides the surveillance results for measured levels of silica in the SFP.

The effect of loss of B^{10} from the Boraflex was evaluated in Attachment 5. The sensitivity calculations were based on the degraded Boraflex base case presented in Attachment 4. The criticality calculations were performed assuming a reduction in either the areal density of B^{10} or a reduction in the areal density of B^{10} with a corresponding reduction in thickness of the Boraflex panel. The most limiting combination that would continue to support the criteria of $K_{eff} < 1.0$ with no soluble boron in the Spent Fuel Pool water was then determined for:

- Region I: The combination was a reduction of both areal density and thickness by a factor of slightly more than two.
- Region II: The combination was no change in thickness and reduction of the areal density by a factor of two.

The effect of silica dissolution on the thickness and areal density of Boraflex panels can be estimated by assuming that dissolution of silica into the pool water is a uniform process. From the data in Attachment 8 it appears that over a period of several years there is only a small decrease in the average panel thickness and areal density given that the average silica concentration has increased at a very slow rate over the elapsed time. Therefore, the assumptions made in the analysis in Attachment 5 are considered very conservative and the resulting K_{eff} is conservatively high.

It is apparent that the amount of Boraflex degradation assumed in the criticality analyses compared to the measured values is greater. Therefore, the calculated K_{eff} would be considered bounding for the present conditions in the SFP.

6.5 Analysis Results

The criticality analyses performed for the Turkey Point spent fuel storage racks show that the acceptance criteria for criticality are met for the storage of Westinghouse 15X15 fuel assemblies under both normal and accident conditions with soluble boron credit, partial credit for the spent fuel rack Boraflex neutron absorber, credit for natural uranium axial blankets (Region I), and the burnup and enrichment limits described above.

The proposed TS 5.6.1.1 changes establish a boron concentration requirement for the water contained in the SFP. Since TS 3/4.9.14, already contains a limit that exceeds this requirement, and soluble boron has always been contained in the SFP water, this requirement will have no effect on normal SFP operations and maintenance. The proposed TS Table 3.9-1 changes establish less restrictive enrichment and burnup requirements for the storage of fuel assemblies. Since the TS for a given enrichment currently contain more restrictive burnup requirements for spent fuel storage, the revised limitations will have no effect on normal pool operations and maintenance.

Based on the results of the criticality analyses presented in Attachment 4, SFP boron concentration of 1100 ppm for accident conditions would maintain the spent fuel storage rack $K_{eff} \leq 0.95$, while compensating for the increased reactivity which could result from a misplaced fuel assembly. The misplaced fuel assembly event bounds a loss of SFP cooling event. The current TS SFP boron concentration limit of 1950 ppm is consistent with the boron concentration normally maintained in the SFP. Since soluble boron has always been contained in the SFP, this requirement will have no effect on normal pool operations and maintenance. The limit of 1950 ppm, bounds the value of 650 ppm for normal conditions, which is required to maintain the spent fuel storage rack $K_{eff} \leq 0.95$ when fuel assemblies are stored in accordance with TS 3/4.9.14.

The 30 day frequency for sampling the boron concentration in the SFP in current TS 3/4.9.14 is adequate because significant reductions in SFP boron concentration result from significant increases in pool volume or significant changes in the sources of non-borated water to the pool. Significant changes in the boron concentration are difficult to produce without detection, since the pool contains such a large volume of water. Soluble boron concentration reduction requires the inflow and outflow of large volumes of water, which are readily detected. Pool inventory changes provide a good indication of potential boron concentration changes. The pool water inventory is monitored by level indication and alarms and by periodic operator rounds of the SFP. Sampling and verification of the SFP boron concentration on a 30-day frequency provides adequate assurance that smaller and less readily identifiable boron concentration reductions are not taking place.

SFP systems, instrumentation, and supporting systems are not modified as a result of the proposed license amendments. Operations involving SFP water cooling and cleanup do not change.

The Turkey Point spent fuel rack criticality analysis also addressed postulated accidents in the SFP. The accidents that can occur in the SFP and their consequences are not significantly affected by taking credit for the soluble boron present in the pool water as a major subcriticality control element.

The criticality analyses confirmed that most SFP accident conditions would not result in an increase in K_{eff} of the spent fuel racks. Examples of such accidents are the drop of a fuel assembly on top of a rack, and the drop or placement of a fuel assembly into the cask loading area. At Turkey Point, the spent fuel assembly rack configuration is such that it precludes the insertion of a fuel assembly in between rack modules. A dropped fuel assembly can only land on the top of the racks.

From a criticality standpoint, the dropped fuel assembly accident assumes a fuel assembly in its most reactive condition is dropped on to the spent fuel racks. The rack structure pertinent for criticality is not excessively deformed. Previous accident analysis with unborated water showed that a dropped fuel assembly which comes to rest horizontally on top of the spent fuel rack has sufficient water separating it from the active fuel height of stored fuel assemblies to preclude neutronic interaction. For the borated water condition, the interaction is even less since the water contains boron, an additional thermal neutron absorber.

However, two accidents can be postulated which could result in an increase in reactivity. The first postulated accident would be a loss of the SFP cooling system. The second would be the misloading of a fuel assembly into a cell for which the restrictions on enrichment and burnup are not satisfied, which can occur in Region II (which bounds all other fuel misloading accidents).

The loss of normal cooling to the SFP water causes an increase in the temperature of the water passing through the stored fuel assemblies. This causes a decrease in water density, which would result in a net increase in reactivity when soluble boron is present in the water and Boraflex neutron absorber panels are present in the racks. A fuel assembly misload accident relates to the use of restricted storage locations based on fuel assembly initial enrichment and burnup. Administrative controls are placed on the loading of assemblies into these locations (Region II). The misloading of a fuel assembly constitutes not meeting the enrichment and burnup requirements of that location. The result of the misloading is to add positive reactivity, increasing K_{eff} toward 0.95.

The amount of soluble boron required to offset each of these postulated accidents was evaluated for all of the storage configurations in the criticality analysis described above. The evaluation established the amount of soluble boron necessary to ensure that the spent fuel rack K_{eff} will be maintained less than or equal to 0.95 should a loss of SFP cooling or a fuel assembly misload accidentoccur.

The amount of soluble boron necessary to mitigate either of these events, 1100 ppm, is bounded by the SFP boron concentration limit contained in TS 3/4.9.14. Based on the double contingency principle, the boron concentration limit for accident conditions contained in the TS does not have to account for both a loss of cooling event and a misload event occurring at the same time.

The radiological consequences of a dropped assembly accident in the SFP do not change because of the presence of soluble boron in the SFP water. The current Updated Final Safety Analysis Report (UFSAR) accident analysis (Section 14.2.1) assumes that a high burnup fuel assembly is dropped, and the outer row of fuel rods (15) in the dropped assembly ruptures releasing the gap radioactive gases. A large fraction of the halogen gases is entrained in the pool water limiting the off-site exposures.

A boron dilution analysis was performed to ensure that sufficient time is available to detect and mitigate dilution of the SFP before the 0.95 K_{eff} design basis is exceeded. The results of the SFP boron dilution analysis are summarized in Attachment 6. Calculations were performed to define the dilution times and volumes for the SFP. The dilution sources available were compiled and evaluated against the calculated dilution volumes, to determine the potential for a SFP boron dilution event.

The analysis shows that a large volume of water (approximately 290,000 gallons) is necessary to dilute the SFP from the TS limit of 1950 ppm to a soluble boron concentration where a K_{eff} of 0.95 would be approached in the SFP. A dilution event large enough to result in a significant reduction in the SFP boron concentration would involve the transfer of a large quantity of water from a dilution source and a significant increase in SFP level which would ultimately overflow the pool. In addition, because of the dilution flow rates available at Turkey Point during normal plant operations (Attachment 6), and the large quantities of water required, any significant dilution of the SFP would only occur over a long period of time (hours to days). Detection of a SFP dilution via level alarms and/or visual inspections would be expected long before a significant dilution would occur. Therefore, it is highly unlikely that any dilution event in the SFP could result in the reduction of the SFP boron concentration to less than the 650 ppm design basis limit.

The proposed TS 5.6.1.1b requires that the spent fuel rack K_{eff} be less than or equal to 0.95 when flooded with water borated to 650 ppm. The dilution analysis (Attachment 6) concluded that large volumes of water are necessary to dilute the SFP water from the 1950 ppm Technical Specification limit to less than the boron concentration limit of 650 ppm. The availability of such large water supplies on site is limited. In addition, the transferability of the available water supplies to the pool is very low due to the small number of possible flow paths, and in many cases impossible due to the physical arrangement of the SFP relative to the supplies.

The SFP dilution analysis assumes thorough mixing of all the nonborated water added to the SFP. It is unlikely, with cooling flow and convection from the spent fuel decay heat, that thorough mixing would not occur. However, if mixing were not adequate, it would be conceivable that a localized pocket of non-borated water could form somewhere in the SFP. This possibility is addressed by the calculation in Attachment 4, which shows that the spent fuel rack K_{eff} will be less than 1.0 on a 95/95 basis with the SFP filled with non-borated water. Thus, even if a pocket of non-borated water formed in the SFP, K_{eff} would not be expected to exceed 1.0 anywhere in the pool.

The Fresh Fuel racks were analyzed (Attachment 7) for the full density water flooding and for introduction of water at optimum density. (Optimum density is defined as the low density of water, which would lead to the highest reactivity of the storage array). Under normal conditions, the fresh fuel racks are maintained in a dry environment. The introduction of water into the fresh fuel rack area is the worst case accident scenario. The water flooding cases analyzed in Attachment 7 are bounding accident situations, which result in the most conservative fuel rack K_{eff}. Since K_{eff} was calculated to be less than 0.95 for the full density analysis and less than 0.98 for the optimum water density moderation analysis, including uncertainties at a 95/95 probability/confidence level, the respective acceptance criteria for criticality are met for storage of the 15x15 fuel assemblies with maximum allowed enrichment.

7.0 Conclusion

The safety analyses in Attachments 4, 5, 6 and 7 demonstrate that the required margin for subcriticality in the SFP and the New Fuel storage racks will not be exceeded.

The combination of the following provides a level of safety:

1. The 95/95 K_{eff} calculation, which shows that the spent fuel rack K_{eff} will remain less than 1.0 when flooded with unborated water.

2. The proposed TS which will ensure that the SFP boron concentration and fuel assembly storage will be maintained consistent with the assumptions in the criticality analysis, thus maintaining the required margin to criticality.

3. The criticality analysis for the Turkey Point spent fuel racks which was performed utilizing the NRC approved Westinghouse Spent Fuel Rack Criticality Analysis Methodology described in WCAP-14416-NP-A, Revision 1, October 25, 1996.

4. The 95/95 K_{eff} calculation, which shows that the New Fuel storage rack K_{eff} will remain less than 0.95 when flooded with unborated water and less than 0.98 for low-density optimum moderation conditions.

In conclusion, the proposed TS amendments provide reasonable assurance of continued protection of the public health and safety.

ATTACHMENT 2

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Introduction

The Nuclear Regulatory Commission has provided standards for determining whether a significant safety hazards consideration exists [10 CFR §50.92(c)]. A proposed amendment to an operating license for a facility involves no significant hazards consideration, if 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. Each standard is discussed below for the proposed amendments.

Discussion

(1) **Operation of the facility in accordance with the proposed amendments would not involve a significant increase in the probability or consequences of an accident previously evaluated.**

There is no increase in the probability of a fuel assembly drop accident in the Spent Fuel Pool (SFP) when considering the presence of soluble boron in the SFP water for criticality control. The handling of the fuel assemblies in the SFP has always been performed in borated water. The consequences of a fuel assembly drop accident in the SFP are not affected when considering the presence of soluble boron.

There is no increase in the probability of the accidental misloading of spent fuel assemblies into the SFP racks when considering the presence of soluble boron in the pool water for criticality control. Fuel assembly placement will continue to be controlled pursuant to approved fuel handling procedures and will be in accordance with the Technical Specification (TS) spent fuel rack storage limitations. There is no increase in the consequences of the accidental misloading of spent fuel assemblies into the SFP racks because criticality analyses demonstrate that the pool will remain subcritical following an accidental misloading if the pool contains an adequate boron concentration. The proposed TS ensure that an adequate SFP boron concentration will be maintained. There is no increase in the probability of the loss of normal cooling to the SFP water when considering the presence of soluble boron in the pool water for subcriticality control since a high concentration of soluble boron has always been maintained in the SFP water.

A loss of normal cooling to the SFP water causes an increase in the temperature of the water passing through the stored fuel assemblies. This causes a decrease in water density, which would result in a net increase in reactivity when soluble boron is present in the water and Boraflex neutron absorber panels are present in the racks. However, the additional negative reactivity provided by the 1950 ppm boron concentration limit, above that provided by the concentration required (650 ppm) to maintain K_{eff} less than or equal to 0.95, will compensate for the increased reactivity which could result from a loss of SFP cooling event. Because adequate soluble boron will be maintained in the SFP water, the consequences of a loss of normal cooling to the SFP will not be increased.

The Fresh Fuel racks are analyzed by employing the "Westinghouse Spent Fuel Rack Criticality Analysis Methodology" approved by the NRC and described in WCAP-14416, NP-A, Revision 1. Only the method for Fresh Fuel storage racks criticality calculations has changed. The method of handling fuel, the maximum fuel enrichment, and the limiting values for criticality have not changed. Therefore, there is no change in the margin of safety for the Fresh Fuel storage racks.

Therefore, based on the conclusions of the above analysis, the proposed changes will not involve a significant increase in the probability or consequences of an accident previously evaluated.

(2) **Operation of the facility in accordance with the proposed amendments would not create the possibility of a new or different kind of accident from any previously evaluated.**

Spent fuel handling accidents are not new or different types of accidents, they have been analyzed in Section 14.2.1 of the Updated Final Safety Analysis Report (UFSAR). Criticality accidents in the SFP are not new or different types of accidents, they have been analyzed in the UFSAR and in the spent fuel storage criticality analysis.

Current TS 3/4.9.14 already contains a limit on the SFP boron concentration. The boron concentration in the SFP has always been maintained near the limit of the RWST boron concentration for refueling purposes. The current TS boron concentration requirement for the SFP water conservatively bounds the boration assumptions of the revised criticality analyses. Since soluble boron has always been maintained in the SFP water, the implementation of this requirement for criticality purposes will have no effect on normal pool operations and maintenance.

Since soluble boron has always been present in the SFP, a dilution of the SFP soluble boron has always been a possibility. However, it was shown in the SFP dilution analysis that a dilution of the Turkey Point SFP which could increase the spent fuel storage rack K_{eff} to greater than 0.95 is not a credible event. Therefore, the implementation of limitations on the SFP boron concentration for criticality purposes will not result in the possibility of a new or different kind of accident.

Proposed TS 3/4.9.14 Table 3.9-1 specifies the requirements for the spent fuel rack storage, which is currently contained in the TS. These proposed new SFP storage limitations are consistent with the assumptions made in the spent fuel rack criticality analysis, and will not have any significant effect on normal SFP operations and maintenance, and will not create any possibility of a new or different kind of accident. Verifications will continue to be performed to ensure that the SFP loading configuration meets specified requirements.

The Fresh Fuel racks are analyzed by employing the "Westinghouse Spent Fuel Rack Criticality Analysis Methodology" approved by the NRC and described in WCAP-14416, NP-A, Revision 1. Only the method for Fresh Fuel storage racks criticality calculations has changed. The method of handling fuel, the maximum fuel enrichment, and the limiting values for criticality have not changed. Therefore, there is no change in the margin of safety for the Fresh Fuel storage racks.

As discussed above, the proposed changes will not create the possibility of a new or different kind of accident from any previously evaluated. There is no significant change in plant configuration, equipment design or equipment.

(3) **Operation of the facility in accordance with the proposed amendments would not involve a significant reduction in a margin of safety.**

The proposed TS changes will provide adequate safety margin to ensure that the stored fuel assembly array will always remain subcritical. Those limits are based on a plant specific criticality analyses performed in accordance with the NRC approved Westinghouse Spent Fuel Rack criticality analysis methodology.

The criticality analysis takes credit for soluble boron to ensure that K_{eff} will be less than or equal to 0.95 under normal circumstances. Storage configurations have been defined using a 95/95 K_{eff} calculation to ensure that the spent fuel rack K_{eff} will be less than 1.0 with no soluble boron. Soluble boron credit is used to provide safety margin by maintaining K_{eff} less than or equal to 0.95, including uncertainties, tolerances, and accident conditions in the presence of SFP soluble boron.

The loss of substantial amounts of soluble boron from the SFP that could lead to exceeding a K_{eff} of 0.95 has been evaluated in the SFP Dilution analysis and shown to be not credible.

The analysis shows that the dilution of the SFP boron concentration from 1950 ppm to 650 ppm is not credible. When this result is combined with the results from the 95/95 criticality analyses, which show that the spent fuel rack K_{eff} will remain less than 1.0 when flooded with unborated water, it provides a level of safety comparable to the conservative criticality analysis methodology required by ANSI 57.2-1983, NUREG-0800, and Regulatory Guide 1.13.

The Fresh Fuel racks are analyzed by employing the "Westinghouse Spent Fuel Rack Criticality Analysis Methodology" approved by the NRC and described in WCAP-14416, NP-A, Revision 1. Only the method for Fresh Fuel storage racks criticality calculations has changed. The method of handling fuel, the maximum fuel enrichment, and the limiting values for criticality have not changed. Therefore, there is no change in the margin of safety for the Fresh Fuel storage racks.

Therefore, the proposed changes in these license amendments will not result in a significant reduction in the plant's margin of safety.

Summary

Based on the reasoning presented above, FPL has determined that the proposed amendments request does not (1) involve a significant increase in the probability or consequences of an accident previously evaluated, (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; therefore the proposed changes do not involve a significant hazards consideration

L-99-176
Attachment 3

ATTACHMENT 3

PROPOSED TECHNICAL SPECIFICATION PAGES

3/4 9-16
5-5

TABLE 3.9-1

SPENT FUEL BURNUP REQUIREMENTS FOR STORAGE IN REGION II OF THE SPENT FUEL PIT

Initial w/o	Discharge Burnup GWD/MT
1.5	0.
1.75	5.0
2.0	9.0
2.2	12.0
2.4	14.8
2.6	17.6
2.8	20.1
3.0	22.6
3.2	25.0
3.4	27.4
3.6	29.6
3.8	31.8
4.0	34.0
4.2	36.1
4.5	39.0

Linear interpolation between two consecutive points will yield conservative results.

Replace with attached

Add

TABLE 3.9-1

SPENT FUEL BURNUP REQUIREMENTS FOR STORAGE IN REGION II OF THE SPENT FUEL PIT

Initial w/o	Discharge Burnup MWD/MTU
1.6	0.0
1.80	3706
2.00	7459
2.20	9724
2.40	12582
2.60	15338
2.63	15914
2.80	17994
3.00	20548
3.25	23312
3.40	25354
3.60	27605
3.88	30256
4.00	31804
4.20	33752
4.40	35599
4.50	36746

Linear interpolation between values may be used for intermediate points.

5.6 FUEL STORAGE

5.6.1 CRITICALITY

a. Keff equivalent to less than 1.0 when flooded with unborated water, which includes a conservative allowance for uncertainties

5.6.1.1 The spent fuel storage racks are designed to provide safe subcritical storage of fuel assemblies by providing sufficient center-to-center spacing or a combination of spacing and ~~poison and shall be maintained~~ with:

b ~~a.~~ A k_{eff} equivalent to less than or equal to 0.95 when flooded with borated to 650 ppm ~~unborated~~ water, which includes a conservative allowance ~~in region 1 of 0.97% Δk/k and in region 2 of 1.96% Δk/k~~ for uncertainties ~~for two-region fuel storage racks.~~ (delete) (Delete)

c. ~~b.~~ A nominal 10.6 inch center-to center distance for Region ~~1~~ I and 9.0 inch center-to-center distance for Region ~~2~~ II for two region fuel storage racks.

d. ~~c.~~ The maximum enrichment loading for fuel assemblies is 4.5 weight percent of U-235.

5.6.1.2 The racks for new fuel storage are designed to store fuel in a safe subcritical array and shall be maintained with:

a. A nominal 21 inch center-to-center spacing to assure k_{eff} equal to or less than 0.98 for optimum moderation conditions and equal to or less than 0.95 for fully flooded conditions.

b. Fuel assemblies placed in the New Fuel Storage Area shall contain no more than 4.5 weight percent of U-235.

ATTACHMENT 4

Criticality Analyses for Fresh and Spent Fuel Storage for Turkey Point Unit 3 and 4 (Degraded Boraflex)

CAB-99-215-Rev. 1

Criticality for Spent Fuel Storage for Turkey Point Units 3 & 4 (Degraded Boraflex)

July 1999

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1.0 Introduction

This report presents the results of criticality analyses of the Florida Power & Light Turkey Point Units 3 and 4 spent fuel storage racks with credit for spent fuel pool soluble boron.

Turkey Point Units 3 and 4 spent fuel pool is divided into two regions. Region 1 is designed to store high reactivity fuel (fresh or low burnup fuel). Region 2 can store lower reactivity (lower enrichment or higher burnup) fuel, but at a higher storage density.

The methodology employed here is contained in the topical report, "Westinghouse Spent Fuel Rack Criticality Analysis Methodology"(1).

The Turkey Point Units 3 and 4 spent fuel racks have been analyzed to allow storage of Westinghouse 15x15 fuel assemblies with nominal enrichments up to 4.5 w/o ^{235}U (with 6 inches long natural uranium axial blanket at top and bottom of the fuel rods) in Region 1 storage cell locations and fuel with an equivalent enrichment up to 1.60 w/o ^{235}U (with no axial blankets) in Region 2. Both analyses take credit for soluble boron in the pool. These analyses include the presence of the Boraflex poison panels in the spent fuel rack. However, the boraflex panels are degraded by shrinkage which creates gaps in the panels.

The analyses require the reactivity of the racks to be below 1.0 with all manufacturing tolerances and uncertainties and without any credit for soluble boron. Credit for soluble boron is taken to provide safety margin by maintaining rack reactivity to be approximately 5% below that for the no boron case, but also less than or equal to 0.95, including uncertainties and tolerances. Credit for soluble boron is also taken under accident conditions.

The following storage configurations and enrichment limits were considered in these analyses:

All Cell Storage in Region 1	Storage of 15x15 fuel assemblies in any cell location. Fuel assemblies must have an initial nominal enrichment no greater than 4.50 w/o ^{235}U with 6 inches long natural uranium axial blanket at top and bottom of the fuel rods. The soluble boron credit required for this storage configuration is 450 ppm.
All Cell Storage in Region 2	Storage of 15x15 fuel assemblies in any cell location. Fuel assemblies must have an initial nominal enrichment no greater than 1.60 w/o ^{235}U or satisfy a minimum burnup requirement for higher initial enrichments. The soluble boron credit required for this storage configuration is 650 ppm.

1.1 Design Description

The Turkey Point Unit 3 spent fuel storage cells for Regions 1 and 2 are shown in Figure 1 on page 26 and Figure 2 on page 27, respectively. These are also used to represent the Turkey Point Unit 4 pool, as they are the same or more limiting. The degraded boraflex panel is shown in Figure 3 on page 28.

The fuel parameters relevant to this analysis are given in Table 1 on page 17. The Table 1 fuel parameters bound the previous fuel designs used in Turkey Point Units 3 & 4. The fuel rod and guide thimble and instrumentation thimble tube cladding are modeled as zircaloy in this analysis. This is conservative with respect to the Westinghouse ZIRLO™ product which is a zirconium alloy containing additional elements including niobium. Niobium has a small absorption cross section which causes more neutron capture in the cladding resulting in a lower reactivity. Therefore, this analysis is conservative with respect to fuel assemblies containing ZIRLO™ cladding in fuel rods and guide thimble and instrumentation thimble tubes.

1.2 Design Criteria

Criticality of fuel assemblies in a fuel storage rack is prevented by the design of the rack which limits fuel assembly interaction. This is done by fixing the minimum separation between fuel assembly and/or placing absorber panels between storage cells.

The design basis for preventing criticality in the spent fuel pool (SFP) or for fresh fuel storage, is examined on the 95/95 basis. The 95/95 basis is defined as the upper limit, with a 95 percent probability at a 95 percent confidence level, of the effective neutron multiplication factor K_{eff} of the fuel assembly array, including uncertainties and manufacturing tolerances.

For the spent fuel pool, the criteria with the credit for soluble boron are:

1. The 95/95 basis K_{eff} for the storage will be less than 1.0, without the presence of soluble boron.

2. The 95/95 basis K_{eff} for the storage will be will be less than or equal to 0.95 with the presence of a defined level of soluble boron in the pool.

These criteria are defined in the topical report, "Westinghouse Spent Fuel Rack Criticality Analysis Methodology"[1].

The design criteria are consistent with ANSI 57.2-1983[2], NRC guidance[3], NUREG-0800[4] and the NRC approved Westinghouse criticality topical report "Westinghouse Spent Fuel Rack Criticality Analysis Methodology"[1].

2.0 Analytical Methods

The criticality calculation method and cross-section values are verified by comparison with critical experiment data for fuel assemblies similar to those for which the racks are designed. This benchmarking data is sufficiently diverse to establish that the method bias and uncertainty will apply to rack conditions which include strong neutron absorbers, large water gaps, low moderator densities and spent fuel pool soluble boron.

The design method which insures the criticality safety of fuel assemblies in the fuel storage rack is described in detail in the Westinghouse Spent Fuel Rack Criticality Analysis Methodology topical report[1]. This report describes the computer codes, benchmarking, and methodology which are used to calculate the criticality safety limits presented in this report for Turkey Point Units 3 and 4.

As determined in the benchmarking in the topical report, the method bias using the described methodology of NITAWL-II, XSDRNPM-S and KENO-Va is 0.00770 ΔK with a 95 percent probability at a 95 percent confidence level uncertainty on the bias of 0.00300 ΔK. These values will be used in this report.

3.0 Criticality Analysis of Region 1 All Cell Spent Fuel Storage

This section describes the analytical techniques and models employed to perform the criticality analysis for the storage of fuel in all cells of the Turkey Point Units 3 and 4 Region 1 spent fuel storage racks with credit for soluble boron.

Section 3.1 describes the no soluble boron K_{eff} calculations. Section 3.2 discusses the results of the spent fuel rack K_{eff} soluble boron credit calculations.

Region 1 fuel storage rack configuration is shown in Figure 1 on page 26 and the rack parameters are shown in Table 2 on page 18.

3.1 No Soluble Boron K_{eff} Calculation

For the no soluble boron conditions the 95/95 basis K_{eff} should be less than 1.0. KENO-Va is used to establish a nominal reference reactivity and PHOENIX-P is used to assess the temperature bias of a normal pool temperature range and the effects of material and construction tolerance variations. A final 95/95 K_{eff} is developed by statistically combining the individual tolerance impacts with the calculational and methodology uncertainties and summing this term with the temperature and method biases and the nominal KENO-Va reference reactivity. The equation for determining the final 95/95 K_{eff} is defined in Reference 1and is shown below:

$$K_{eff} = K_{nominal} + B_{method} + B_{temp} + B_{self} + B_{uncert}$$

where:

$K_{nominal}$	=	nominal conditions KENO-Va K_{eff}
B_{method}	=	method bias determined from benchmark critical comparisons
B_{temp}	=	temperature bias
B_{self}	=	^{10}B self shielding bias, if applicable.
B_{uncert}	=	statistical summation of uncertainty components
		= [Sum{tolerance(i)^2 ... or ... uncertainty(i)}^{2}]**1/2

Following assumptions were used to develop the nominal KENO-Va model for storage of fuel assemblies in all cells of the Turkey Point Units 3 and 4 spent fuel storage rack:

1. The fuel assembly parameters relevant to the criticality analysis were based on the Westinghouse 15x15 DRFA fuel design with nominal enrichment of 4.5 w/o and with 6 inch natural uranium axial blanket at top and bottom (see Tables 1 and 2 for fuel parameters).
2. The fuel pellets were modeled assuming nominal values for pellet density and dishing fraction.
3. No credit was taken for any ^{234}U or ^{236}U in the fuel, nor was any credit taken for the buildup of fission product poison material.
4. No credit was taken for any grids in the assembly.
5. No credit was taken for any burnable absorber in the fuel rods.
6. Credit was taken for the presence of spent fuel rack Boraflex poison panels. ^{10}B areal density of 0.020 gm/cm^2 was used.
7. The moderator was water with 0 ppm soluble boron at a temperature of 68°F. A water density of 1.0 gm/cm^3 was used.
8. The array was infinite in lateral (x and y) extent and finite in axial (vertical) extent.
9. All available storage cells were loaded with fuel assemblies.

Fuel assembly data is shown in Table 1 on page 17. Dimensions for the Region 1 fuel racks are shown in Table 2 on page 18 and the cell arrangement is shown in Figure 1 on page 26.

With the above assumptions, the nominal KENO-Va calculation resulted in a K_{eff} of 0.93970 ± 0.00081 under normal conditions.

Following biases are included:

Methodology: The benchmarking bias as determined for the Westinghouse KENO-Va methodology was considered.

Water Temperature: A reactivity bias determined in PHOENIX-P was applied to account for the effect of the normal range of spent fuel pool water temperatures (50°F to 185°F).

Particle Size Effect in Boraflex: A bias on reactivity resulting from the finite particle size of the boron bearing compound in Boraflex panels.

All biases are shown in Table 3 on page 19. To evaluate the reactivity effects of possible variations in material characteristics and mechanical/construction dimensions, PHOENIX-P perturbation calculations were performed. For the Turkey Point Units 3 and 4 spent fuel rack all cell storage configuration, UO_2 material tolerances were considered along with construction tolerances. Uncertainties associated with calculation and methodology accuracy were also included in the statistical summation of uncertainty components. Nominal values are shown in Table 2 on page 18 and manufacturing tolerances and uncertainties are shown in Table 3 on page 19.

Following tolerance and uncertainty components were considered in the total uncertainty statistical summation:

^{235}U Enrichment: The standard DOE enrichment tolerance of ±0.05 w/o ^{235}U about the nominal reference enrichment of 4.50 w/o ^{235}U was considered.

UO_2 Density: A ±2.0% variation about the nominal reference theoretical density was considered.

Fuel Pellet Dishing: A variation in fuel pellet dishing fraction from 0.0% to the nominal dishing was considered.

Storage Cell I.D.: The +0.050/-0.025 inch tolerance about the nominal 8.75 inch reference cell I.D. was considered.

Storage Cell Pitch: The +0.12/–0.12 inch tolerance about the nominal 10.60 inch reference cell pitch was considered.

Stainless Steel Wall Thickness: The +0.007/ -0.007 inch tolerance about the nominal 0.075 inch reference stainless steel wall thickness was considered.

Boraflex Thickness: A tolerance of +0.007/-0.007 inch about the reference 0.078 inch thickness of Boraflex is included.

Boraflex Width: A manufacturing tolerance of +0.075/-0.075 inch around the nominal Boraflex width of 7.5 inches is included.

Boraflex Length: Nominal Boraflex length is 139.4 inches with a manufacturing uncertainty of +0.25/-0.25 inch. Based on a previous study, the reactivity variation due to this small tolerance is negligible. No calculation was performed here for this tolerance.

Wrapper Thickness: A manufacturing tolerance of +0.002/-0.002 inch around the nominal stainless steel wrapper thickness of 0.02 inch is included.

Poison Cavity: The +0.010/-0.010 inch tolerance around the nominal poison cavity thickness of 0.090 inch is included.

Assembly Position: The KENO-Va reference reactivity calculation assumed fuel assemblies were symmetrically positioned (centered) within the storage cells. Potentially an increase in reactivity can occur if the corners of the four fuel assemblies were positioned together.

Calculation Uncertainty: The 95 percent probability/95 percent confidence level uncertainty on the KENO-Va nominal reference K_{eff} was considered.

Methodology Uncertainty: The 95 percent probability/95 percent confidence uncertainty in the benchmarking bias as determined for the Westinghouse KENO-Va methodology was included (0.00300 ΔK).

These manufacturing tolerances and uncertainties are convoluted and added to the nominal K_{eff} to arrive at the final K_{eff} on a 95/95 basis.

This summation is shown in Table 3 on page 19 and results in a 95/95 basis K_{eff} of 0.96150.

Since K_{eff} is less than 1.0. the Turkey Point Units 3 and 4 spent fuel racks will remain subcritical under conditions when all cells are loaded with 4.50 w/o ^{235}U 15x15 fuel assemblies with natural uranium axial blankets and no soluble boron is present in the spent fuel pool water. This meets the design basis.

3.2 Soluble Boron Credit K_{eff} Calculations

The criterion for defining the boron level, used here is that the K_{eff} of the racks should be approximately 5% below the value of K_{eff} for no boron and also below or equal to 0.95. Based on the results of Section 3.1, the target 95/95 K_{eff} is approximately 0.912.

To determine the amount of soluble boron required to maintain the target K_{eff}, KENO-Va was used to establish a nominal reference reactivity and PHOENIX-P was used to assess the pool temperature bias, the effects of material and construction tolerances, as described in Section 3.1.

The assumptions used to develop the nominal KENO-Va model for soluble boron credit for all cell storage in the Turkey Point Units 3 and 4 spent fuel racks were similar to those in Section 3.1 except for assumption 7 regarding the moderator soluble boron concentration. The moderator was replaced with water containing 450 ppm soluble boron.

With the above assumptions, the KENO-Va calculation for the nominal case with 450 ppm soluble boron in the moderator resulted in a K_{eff} of 0.86326 $\pm$ 0.00076.

All biases and tolerances are shown in Table 4 on page 20. A final 95/95 K_{eff} was developed by statistical convolution of the individual tolerance impacts and the calculational and methodology uncertainties and then summing this term with the biases and the nominal KENO-Va reference reactivity. The same equation as defined in Section 3.1 is used for the final 95/95 K_{eff}. Summation shown in Table 4 on page 20, results in a 95/95 basis K_{eff} of 0.90391.

Since K_{eff} is less than or equal to 0.95 including soluble boron credit and uncertainties at a 95/95 probability/confidence level, the acceptance criteria for criticality is met for all cell storage of 15x15 fuel assemblies in the Turkey Point Units 3 and 4 spent fuel racks. Storage of fuel assemblies with nominal enrichments no greater than 4.50 w/o ^{235}U with natural uranium axial blankets is acceptable for storage in all cells with the presence of 450 ppm soluble boron.

4.0 Criticality Analysis of Region 2 All Cell Spent Fuel Storage

This section describes the analytical techniques and models employed to perform the criticality analysis and reactivity equivalencing evaluations for the storage of fuel in all cells of the Turkey Point Units 3 and 4 Region 2 spent fuel storage racks with credit for soluble boron.

Section 4.1 describes the no soluble boron K_{eff} calculations. Section 4.2 discusses the results of the spent fuel rack K_{eff} soluble boron credit calculations. Finally, Section 4.3 presents the results of calculations performed to show the minimum burnup requirements for assemblies with initial enrichments above those determined in Section 4.1.

4.1 No Soluble Boron K_{eff} Calculation

For the no soluble boron conditions the 95/95 basis K_{eff} should be less than 1.0. KENO-Va is used to establish a nominal reference reactivity and PHOENIX-P is used to assess the temperature bias of a normal pool temperature range and the effects of material and construction tolerance variations. A final 95/95 K_{eff} is developed by statistically combining the individual tolerance impacts with the calculational and methodology uncertainties and summing this term with the temperature and method biases and the nominal KENO-Va reference reactivity. The equation for determining the final 95/95 K_{eff} is defined in Reference 1 and is shown below:

$$K_{eff} = K_{nominal} + B_{method} + B_{temp} + B_{self} + B_{uncert}$$

where:

$K_{nominal}$ = nominal conditions KENO-Va K_{eff}

B_{method} = method bias determined from benchmark critical comparisons

B_{temp} = temperature bias

B_{self} = ^{10}B self shielding bias, if applicable.

B_{uncert} = statistical summation of uncertainty components

= [Sum{tolerance(i)^2 ... or ... uncertainty(i)}^{2}]**1/2

Following assumptions were used to develop the no soluble boron nominal KENO-Va model for storage of fuel assemblies in all cells of the Turkey Point Units 3 and 4 spent fuel storage racks:

1. The fuel assembly parameters relevant to the criticality analysis were based on the Westinghouse 15x15 DRFA fuel design (see Table 1 on page 17 for fuel parameters).
2. Fuel assemblies contain uranium dioxide at a nominal enrichment of 1.60 w/o ^{235}U over the entire length of each rod.
3. The fuel pellets were modeled assuming nominal values for theoretical density and dishing fraction.
4. No credit was taken for any natural or reduced enrichment axial blankets.
5. No credit was taken for any ^{234}U or ^{236}U in the fuel, nor was any credit taken for the buildup of fission product poison material.
6. No credit was taken for any grids in the assembly.
7. No credit was taken for any burnable absorber in the fuel rods.
8. Credit was taken for the presence of spent fuel rack Boraflex poison panels. ^{10}B areal density of 0.012 gm/cm^2 was used.
9. The moderator was water with 0 ppm soluble boron at a temperature of 68°F. A water density of 1.0 gm/cm^3 was used.
10. The array was infinite in lateral (x and y) extent and finite in axial (vertical) extent.
11. Fuel storage cells were loaded with fuel assemblies in all cell arrangement.

Fuel assembly data is shown in Table 1 on page 17. Dimensions for the Region 2 fuel racks are shown in Table 5 on page 21 and the cell arrangement is shown in Figure 2 on page 27.

With the above assumptions, the nominal KENO-Va calculation resulted in a K_{eff} of 0.94785 $\pm$ 0.00042 under normal conditions.

Following biases were included:

Methodology: The benchmarking bias as determined for the Westinghouse KENO-Va methodology was considered.

Water Temperature: A reactivity bias determined in PHOENIX-P was applied to account for the effect of the normal range of spent fuel pool water temperatures (50°F to 185°F).

Particle Size Effect in Boraflex: A bias on reactivity resulting from the finite particle size of the boron bearing compound in Boraflex panels.

All biases are shown in Table 6 on page 22. To evaluate the reactivity effects of possible variations in material characteristics and mechanical/construction dimensions, PHOENIX-P perturbation calculations were performed. For the Turkey Point Units 3 and 4 spent fuel rack all configuration, UO_2 material tolerances were considered along with construction tolerances. Uncertainties associated with calculation and methodology accuracy were also considered in the statistical summation of uncertainty components. Nominal values are shown in Table 5 on page 21 and manufacturing tolerances and uncertainties are shown in Table 6 on page 22.

The following tolerance and uncertainty components were considered in the total uncertainty statistical summation:

^{235}U Enrichment: The standard DOE enrichment tolerance of ±0.05 w/o ^{235}U about the nominal reference enrichment of 1.60 w/o ^{235}U was considered.

UO_2 Density: A ±2.0% variation about the nominal reference theoretical density was considered.

Fuel Pellet Dishing: A variation in fuel pellet dishing fraction from 0.0% to the nominal dishing.

Storage Cell I.D.: The +0.025/-0.025 inch tolerance about the nominal 8.80 inch reference cell I.D. was considered.

Storage Cell Pitch: The +0.07/–0.03 inch tolerance about the nominal 9.0 inch reference cell pitch was considered.

Stainless Steel Wall Thickness: The +0.007/-0.007 inch tolerance about the nominal 0.075 inch reference stainless steel wall thickness was considered.

Boraflex Thickness: A tolerance of +0.007/-0.007 inch about the reference 0.051 inch thickness of Boraflex is included.

Boraflex Width: A manufacturing tolerance of +0.075/-0.075 inch around the nominal Boraflex width of 7.5 inches is included.

Boraflex Length: Nominal Boraflex length is 139.4 inches with a manufacturing uncertainty of +0.25/-0.25 inch. Based on a previous study, the reactivity variation due to this small tolerance is negligible. No calculation was performed here for this tolerance.

Wrapper Thickness: A manufacturing tolerance of +0.002/-0.002 inch around the nominal stainless steel wrapper thickness of 0.02 inch is included.

Assembly Position: The KENO-Va reference reactivity calculation assumed fuel assemblies were symmetrically positioned (centered) within the storage cells. Potentially an increase in reactivity can occur if the corners of the four fuel assemblies were positioned together.

Calculation Uncertainty: The 95 percent probability/95 percent confidence level uncertainty on the KENO-Va nominal reference K_{eff} was considered.

Methodology Uncertainty: The 95 percent probability/95 percent confidence uncertainty in the benchmarking bias as determined for the Westinghouse KENO-Va methodology was considered.

These manufacturing tolerances and uncertainties are convoluted and added to the nominal K_{eff} to arrive at the final K_{eff} on a 95/95 basis.

This summation is shown in Table 6 on page 22 and results in a 95/95 basis K_{eff} of 0.97217.

Since K_{eff} is less than 1.0, the Turkey Point Units 3 and 4 spent fuel racks will remain subcritical when all cells are loaded with nominal 1.60 w/o ^{235}U 15x15 fuel assemblies and no soluble boron is present in the spent fuel pool water. This meets the design basis.

4.2 Soluble Boron Credit K_{eff} Calculations

The criterion for defining the boron level, used here is that the K_{eff} of the racks should be approximately 5% below the value of K_{eff} for no boron and also below or equal to 0.95. Based on the results of Section 4.1, the target K_{eff} is approximately 0.922.

To determine the amount of soluble boron required to maintain the target K_{eff}, KENO-Va was used to establish a nominal reference reactivity and PHOENIX-P was used to assess the pool temperature bias, the effects of material and construction tolerances, as described in Section 4.1.

The assumptions used to develop the nominal KENO-Va model for soluble boron credit for all cell storage in the Turkey Point Units 3 and 4 spent fuel racks were similar to those in Section 4.1 except for assumption 9 regarding the moderator soluble boron concentration. The moderator was replaced with water containing 250 ppm soluble boron.

With the above assumptions, the KENO-Va calculation for the nominal case with 250 ppm soluble boron in the moderator resulted in a K_{eff} of 0.89088 $\pm$ 0.00040.

All biases and tolerances are shown in Table 7 on page 23. A final 95/95 K_{eff} was developed by statistical convolution of the individual tolerance impacts and the calculational and methodology uncertainties and then summing this term with the biases and the nominal KENO-Va reference reactivity. The same equation as defined in Section 4.1 is used for the final 95/95 K_{eff}. The summation is shown in Table 7 on page 23 and results in a 95/95 basis K_{eff} of 0.91663.

Since K_{eff} is less than or equal to 0.95 including soluble boron credit and uncertainties at a 95/95 probability/confidence level, the acceptance criteria for criticality is met for all cell storage of 15x15 fuel assemblies with nominal enrichments no greater than 1.60 w/o ^{235}U in the Turkey Point Units 3 and 4 spent fuel racks with the presence of 250 ppm soluble boron.

4.3 Burnup Credit Reactivity Equivalencing

Storage of fuel assemblies with initial enrichments higher than 1.60 w/o ^{235}U in all cells of the Turkey Point Units 3 and 4 spent fuel racks is achievable by means of burnup credit using reactivity equivalencing. The concept of reactivity equivalencing is based on the reactivity decrease associated with fuel depletion. For burnup credit, a series of reactivity calculations is performed to generate a set of enrichment-fuel assembly discharge burnup ordered pairs which all yield an equivalent K_{eff} when stored in the spent fuel storage racks.

Figure 4 on page 29 shows the constant K_{eff} contours generated for all cell storage in the Turkey Point Units 3 and 4 spent fuel racks. This curve represents combinations of fuel enrichment and discharge burnup which yield the same rack multiplication factor (K_{eff}) as the rack loaded with 1.60 w/o ^{235}U fuel assemblies at zero burnup in all cell locations.

Uncertainties associated with burnup credit include a reactivity uncertainty of 0.01 ΔK at 30,000 MWD/MTU applied linearly to the burnup credit requirement to account for calculation and depletion uncertainties and 5% on the calculated burnup to account for burnup measurement

uncertainty. The amount of additional soluble boron needed to account for these uncertainties in the burnup requirement of Figure 4 on page 29 was 400 ppm. This is additional boron above the 250 ppm required in Section 4.2. This results in a total soluble boron requirement of 650 ppm.

It is important to recognize that the curve in Figure 4 on page 29 is based on calculations of constant rack reactivity. In this way, the environment of the storage rack and its influence on assembly reactivity is implicitly considered. For convenience, the data from Figure 4 on page 29 are also provided in Table 9 on page 25. Use of linear interpolation between the tabulated values is acceptable since the curve shown in Figure 4 on page 29 is linear in between the tabulated points.

The effect of axial burnup distribution on assembly reactivity has been considered in the development of the Turkey Point Units 3 and 4 all cell storage burnup credit limit. Previous evaluations have been performed to quantify axial burnup reactivity effects and to confirm that the reactivity equivalencing methodology described in Reference 1 results in calculations of conservative burnup credit limits. The evaluations show that axial burnup effects only become important at burnup-enrichment combinations which are above those calculated for the Turkey Point Units 3 and 4 all cell storage burnup credit limit. Therefore, additional accounting of axial burnup distribution effects in the Turkey Point Units 3 and 4 all cell storage burnup credit limit is not necessary.

5.0 Discussion of Postulated Accidents in the Spent Fuel Pool

Most accident conditions will not result in an increase in K_{eff} of the rack. Examples are:

Fuel assembly drop on top of rack	The rack structure pertinent for criticality is not excessively deformed and the dropped assembly which comes to rest horizontally on top of the rack has sufficient water separating it from the active fuel height of stored assemblies to preclude neutronic interaction.
Fuel assembly drop between rack modules or between rack modules and spent fuel pool wall	The design of the spent fuel racks and fuel handling equipment is such that it precludes the insertion of a fuel assembly between rack modules. However, it is possible that a fuel assembly can be positioned between the rack modules and the spent fuel pool wall. The reactivity increase caused by this incident is bounded by the misplacement of a fuel assembly inside the spent fuel racks where it does not meet the enrichment or burnup restrictions.

However, two accidents can be postulated for each storage configuration which can increase reactivity beyond the analyzed condition. The first postulated accident would be a change in the spent fuel pool water temperature and the second would be a misload of an assembly into a cell for which the restrictions on enrichment or burnup are not satisfied.

Loss of Cooling Accident

For the change in spent fuel pool water temperature accident (due to loss of cooling), a temperature range of 32°F to 240°F is considered. Calculations were performed for all Turkey Point Units 3 and 4 storage configurations to determine the reactivity change caused by a change in the spent fuel pool water temperature outside the normal range (50°F to 185°F). The results of these calculations are tabulated in Table 8 on page 24 for both Regions 1 and 2.

Assembly Misloaded Accident

The misloaded assembly accident addresses the largest reactivity increase caused by a 4.50 w/o enriched 15x15 unirradiated fuel assembly misplaced into a storage cell for which the restrictions on enrichment or burnup are not satisfied. For Region 1, there are no restrictions on assembly burnup for placing the assemblies in an all cell configuration. Consequently, this accident is not relevant to Region 1. For Region 2, the accidental placement of a high reactivity assembly (4.5% enrichment and no burnup) in the all cell configuration can lead to a reactivity increase.The results of these calculations are also tabulated in Table 8. It shows the increase in the soluble boron level required to assure that with this accidental misload, the target reactivity with boron is still met.

Boron Dilution

Further it has been shown in a companion report that the time and quantities of water to dilute the boron in the pool to the normal boron levels meet the design criterion.

For an occurrence of the above postulated accident conditions, the double contingency principle of ANSI/ANS-8.1-1983[5] can be applied. It specifies that assumption of two unlikely, independent, concurrent events need not be considered to ensure protection against a criticality accident. Dilution of boron in water and misload of an assembly are two independent accidents.Thus, for these postulated accident conditions, the presence of additional soluble boron in the storage pool water (above the concentration required for normal conditions and reactivity equivalencing) can be assumed as a realistic initial condition.

Based on the above discussion, should a loss of spent fuel pool cooling accident or a fuel assembly misload occur in the Turkey Point Units 3 and 4 spent fuel racks, K_{eff} will be maintained below the target level as well as less than or equal to 0.95 with the presence of at least 1100 ppm of soluble boron in the spent fuel pool.

6.0 Soluble Boron Credit Summary

Spent fuel pool soluble boron has been used in this criticality analysis to offset storage rack and fuel assembly tolerances, calculational uncertainties, uncertainty associated with reactivity equivalencing (burnup credit) and the reactivity increase caused by postulated accident conditions. The total soluble boron concentration required to be maintained in the spent fuel pool is a summation of each of these components. Table 8 on page 24 summarizes the storage configurations and corresponding soluble boron credit requirements.

7.0 Summary of Criticality Results

For the storage of Westinghouse 15x15 fuel assemblies in the Turkey Point Units 3 and 4 spent fuel storage racks, the acceptance criteria for criticality requires the effective neutron multiplication factor, K_{eff}, to be < 1.0 with all tolerances and uncertainties with no soluble boron, and $\leq$ 0.95 including uncertainties, tolerances and accident conditions in the presence of spent fuel pool soluble boron.

This report shows that the acceptance criteria for criticality is met for the Turkey Point Units 3 and 4 spent fuel racks for the storage of Westinghouse 15x15 fuel assemblies under both normal and accident conditions with enrichment limits tabulated below.

<u>Enrichment Limits</u>

All Cell Storage in Spent Fuel Pool in Region 1	Storage of 15x15 fuel assemblies in any cell location. Fuel assemblies must have an initial nominal enrichment no greater than 4.50 w/o ^{235}U and have 6 inches of natural uranium blanket at top and bottom. The soluble boron credit required for this storage configuration is 450 ppm.
All Cell Storage in Spent Fuel Pool in Region 2	Storage of 15x15 fuel assemblies in any cell location. Fuel assemblies must have an initial nominal enrichment no greater than 1.6 w/o or satisfy a minimum burnup requirement for higher initial enrichments. The soluble boron credit required for this storage configuration is 650 ppm.

Soluble boron level needed in the spent fuel pool for normal and accident conditions are shown in Table 8 on page 24.

The analytical methods employed herein conform to ANSI N18.2-1973, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants", Section 5.7 Fuel Handling System; ANSI 57.2-1983, "American Nuclear Society, American National Standard Design Requirements for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Plants", Section 6.4.2; ANSI 8.1-1983, "Validation of Calculational Methods for Nuclear Criticality Safety"; the NRC Standard Review Plan[4], Section 9; and the NRC approved Westinghouse criticality topical report "Westinghouse Spent Fuel Rack Criticality Analysis Methodology"[1].

Table 1. Nominal Fuel Parameters Employed in the Criticality Analysis

Parameter	Westinghouse 15x15 DRFA
Number of Fuel Rods per Assembly	204
Rod Clad O.D. (inch)	0.422
Clad Thickness (inch)	0.0243
Fuel Pellet O.D. (inch)	0.3659
Fuel Pellet Density (% of Theoretical)	96
Fuel Pellet Dishing Factor (%)	1.187
Rod Pitch (inch)	0.563
Number of Zirc Guide Tubes	20
Guide Tube O.D. (inch)	0.533
Guide Tube Thickness (inch)	0.017
Number of Instrument Tubes	1
Instrument Tube O.D. (inch)	0.533
Instrument Tube Thickness (inch)	0.017

Table 2. Turkey Point 3 and 4 Region 1 Spent Fuel Storage Cell and Fuel Nominal Parameters

Rack Parameter	Unit	Value
Rack Cell Inner Dimension	inch	8.75
Rack Cell Pitch	inch	10.60
Rack Material		SS
Rack Wall Thickness	inch	0.075
Wrapper Material		SS
Wrapper Plate Thickness	inch	0.020
Poison Panel Thickness	inch	0.078
Poison Cavity Thickness	inch	0.090
Poison Panel Width	inch	7.5
Poison Panel Length	inch	139.4
^{10}B Loading in Boraflex	gm/cm^2	0.020
Bottom of Boraflex Above Support Pad	inch	6.16
Fuel Parameter		
Axial Blanket at Each End of the Rods	inch	6.0
Blanket Material		Natural Uranium
Nominal Fuel Enrichment of the Central Length	%	4.5
Blanket Region Enrichment	%	0.79
Theoretical Density	%	96

Table 3. Region 1 - No Soluble Boron

Base Keno Reference Reactivity		0.93970
Calculation and Methodology Biases	**Range**	
Methodology (Benchmark) Bias		0.00770
Pool Temperature Bias	50 F to 185 F	0.00111
Boron Particles in Boraflex		0.00140
Total Bias		0.01021
Tolerances and Uncertainties	**Parameter Variation**	**Reactivity Variation**
Fuel Enrichment	+0.05/-0.05 %	0.00184
Fuel Density	+2/-2 %	0.00255
Fuel Pellet Dishing	-1.187 %	0.00148
Rack Cell Inner Dimension	+0.05/-0.025 inch	0.00204
Rack Cell Pitch	+0.12/-0.12 inch	0.00950
Rack Wall Thickness	+0.007/-0.007 inch	0.00031
Wrapper Plate Thickness	+0.002/-0.002 inch	0.00000
Poison Panel Thickness	+0.007/-0.007 inch	0.00251
Poison Cavity Thickness	+0.010/-0.010 inch	0.00001
Poison Panel Width	+0.075/-0.075 inch	0.00020
Asymmetric Assembly Position		0.00325
Calculation Uncertainty		0.00134
Benchmark Bias Uncertainty		0.00300
Total Uncertainty (convoluted)		0.01159
Final K_{eff} on 95/95 Basis		0.96150

Table 4. Region 1 - With Soluble Boron (450 ppm)

Base Keno Reference Reactivity		0.88326
Calculation and Methodology Biases	**Range**	
Methodology (Benchmark) Bias		0.00770
Pool Temperature Bias	50 F to 185 F	0.00084
Boron Particles in Boraflex		0.00140
Total Bias		0.00994
Tolerances and Uncertainties	**Parameter Variation**	**Reactivity Variation**
Fuel Enrichment	+0.05/-0.05 %	0.00206
Fuel Density	+2/-2 %	0.00312
Fuel Pellet Dishing	-1.187 %	0.00181
Rack Cell Inner Dimension	+0.05/-0.025 inch	0.00169
Rack Cell Pitch	-0.12 inch	0.00886
Rack Wall Thickness	+0.007/-0.007 inch	0.00014
Wrapper Plate Thickness	+0.002/-0.002 inch	0.00000
Poison Panel Thickness	+0.007/-0.007 inch	0.00228
Poison Cavity Thickness	+0.010/-0.010 inch	0.00001
Poison Panel Width	+0.075/-0.075 inch	0.00018
Asymmetric Assembly Position		0.00045
Calculation Uncertainty		0.00125
Benchmark Bias Uncertainty		0.00300
Total Uncertainty (convoluted)		0.01071
Final K_{eff} on 95/95 Basis		**0.90391**

Table 5. Turkey Point 3 and 4 Region 2 Spent Fuel Storage Cell Nominal Parameters

Parameter	Unit	Value
Rack Cell Inner Dimension	inch	8.80
Rack Cell Pitch	inch	9.0
Rack Material		SS
Rack Wall Thickness	inch	0.075
Wrapper Material		SS
Wrapper Plate Thickness	inch	0.020
Poison Panel Thickness	inch	0.051
Poison Cavity Thickness	inch	0.090
Poison Panel Width	inch	7.5
Poison Panel Length	inch	139.4
^{10}B Loading in Boraflex	gm/cm^2	0.012
Bottom of Boraflex Above Support Pad	inch	6.16

Table 6. Region 2 - No Soluble Boron

Base Keno Reference Reactivity		0.94785
Calculation and Methodology Biases	**Range**	
Methodology (Benchmark) Bias		0.00770
Pool Temperature Bias	50 F to 185 F	0.00140
Boron Particles in Boraflex		0.00317
Total Bias		0.01227
Tolerances and Uncertainties	**Parameter Variation**	**Reactivity Variation**
Fuel Enrichment	+0.05/-0.05 %	0.00959
Fuel Density	+2/-2 %	0.00272
Fuel Pellet Dishing	-1.187 %	0.00134
Rack Cell Inner Dimension	+0.025/-0.025 inch	0.00000
Rack Cell Pitch	+0.07/-0.03 inch	0.00092
Rack Wall Thickness	+0.007/-0.007 inch	0.00000
Wrapper Plate Thickness	+0.002/-0.002 inch	0.00000
Poison Panel Thickness	+0.007/-0.007 inch	0.00579
Poison Cavity Thickness	+0.010/-0.010 inch	0.00000
Poison Panel Width	+0.075/-0.075 inch	0.00055
Asymmetric Assembly Position	–	0.00000
Calculation Uncertainty		0.00069
Benchmark Bias Uncertainty		0.00300
Total Uncertainty (convoluted)		0.01205
Final K_{eff} on 95/95 Basis		**0.97217**

Table 7. Region 2 - With Soluble Boron (250 ppm)

Base Keno Reference Reactivity		0.89088
Calculation and Methodology Biases	**Range**	
Methodology (Benchmark) Bias		0.00770
Pool Temperature Bias	50 F to 185 F	0.00164
Boron Particles in Boraflex		0.00317
Total Bias		0.01251
Tolerances and Uncertainties	**Parameter Variation**	**Reactivity Variation**
Fuel Enrichment	+0.05/-0.05 %	0.01037
Fuel Density	+2/-2 %	0.00383
Fuel Pellet Dishing	-1.187 %	0.00223
Rack Cell Inner Dimension	+0.025/-0.025 inch	0.00015
Rack Cell Pitch	+0.07/-0.03 inch	0.00158
Rack Wall Thickness	+0.007/-0.007 inch	0.00020
Wrapper Plate Thickness	+0.002/-0.002 inch	0.00001
Poison Panel Thickness	+0.007/-0.007 inch	0.00592
Poison Cavity Thickness	+0.010/-0.010 inch	0.00004
Poison Panel Width	+0.075/-0.075 inch	0.00102
Asymmetric Assembly Position	+0.07/-0.03 inch	0.00000
Calculation Uncertainty		0.00066
Benchmark Bias Uncertainty		0.00300
Total Uncertainty (convoluted)		0.01324
Final K_{eff} on 95/95 Basis		**0.91663**

Table 8. Spent Fuel Pool Soluble Boron Requirements

Region	Normal (ppm)	For Loss of Cooling (ppm)	For Misloaded Assembly (ppm)	Soluble Boron for Accidents (ppm)
1	450	50	0	500
2	650	< 450	450	1100
Pool	650			1100

Table 9. Burnup Needed for Fuel Storage in Region 2 of the Spent Fuel Pool

Enrichment (%)	Burnup (MWD/MTU)
1.60	0.0
1.80	3706
2.00	7459
2.20	9724
2.40	12582
2.60	15338
2.63	15914
2.80	17994
3.00	20548
3.25	23312
3.40	25354
3.60	27605
3.88	30256
4.00	31804
4.20	33752
4.40	35599
4.50	36746

Figure 1. Turkey Point Units 3 and 4 Region 1 Spent Fuel Pool Storage Cell Nominal Dimensions

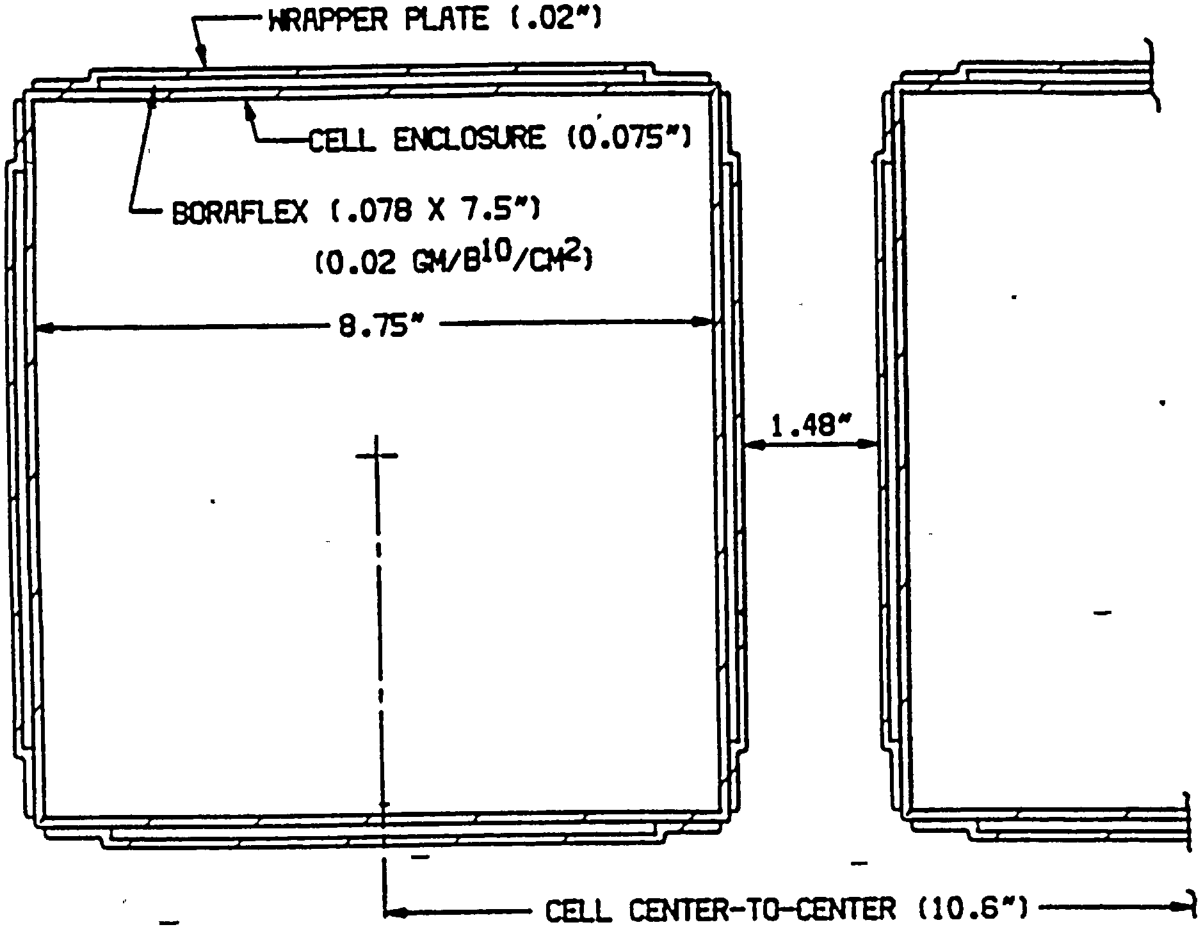


Figure 2. Turkey Point Units 3 and 4 High Density Spent Fuel Pool Storage Cell (Region 2) Nominal Dimensions

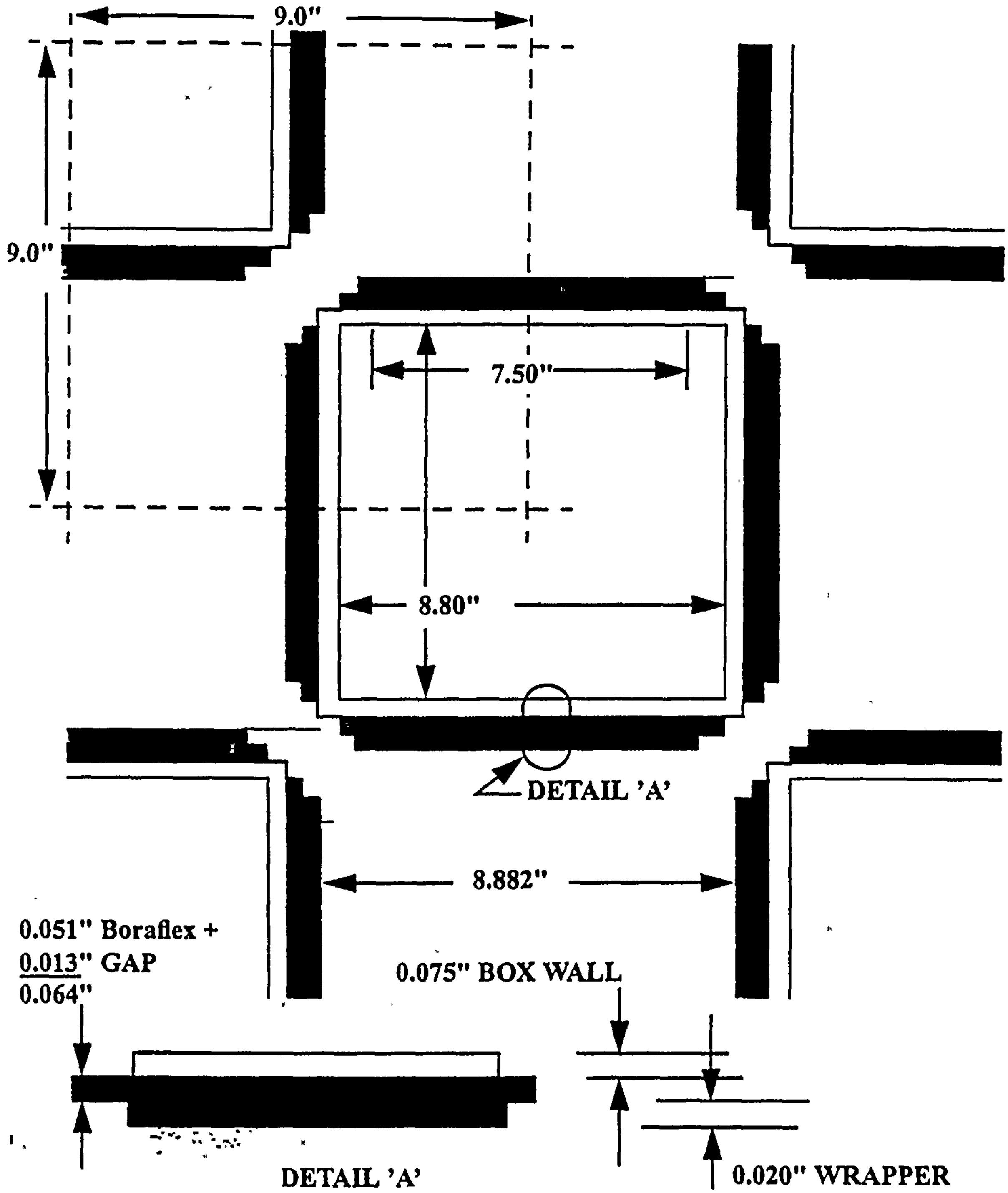


Figure 3. Degraded Boraflex Panel
(Not to Scale)

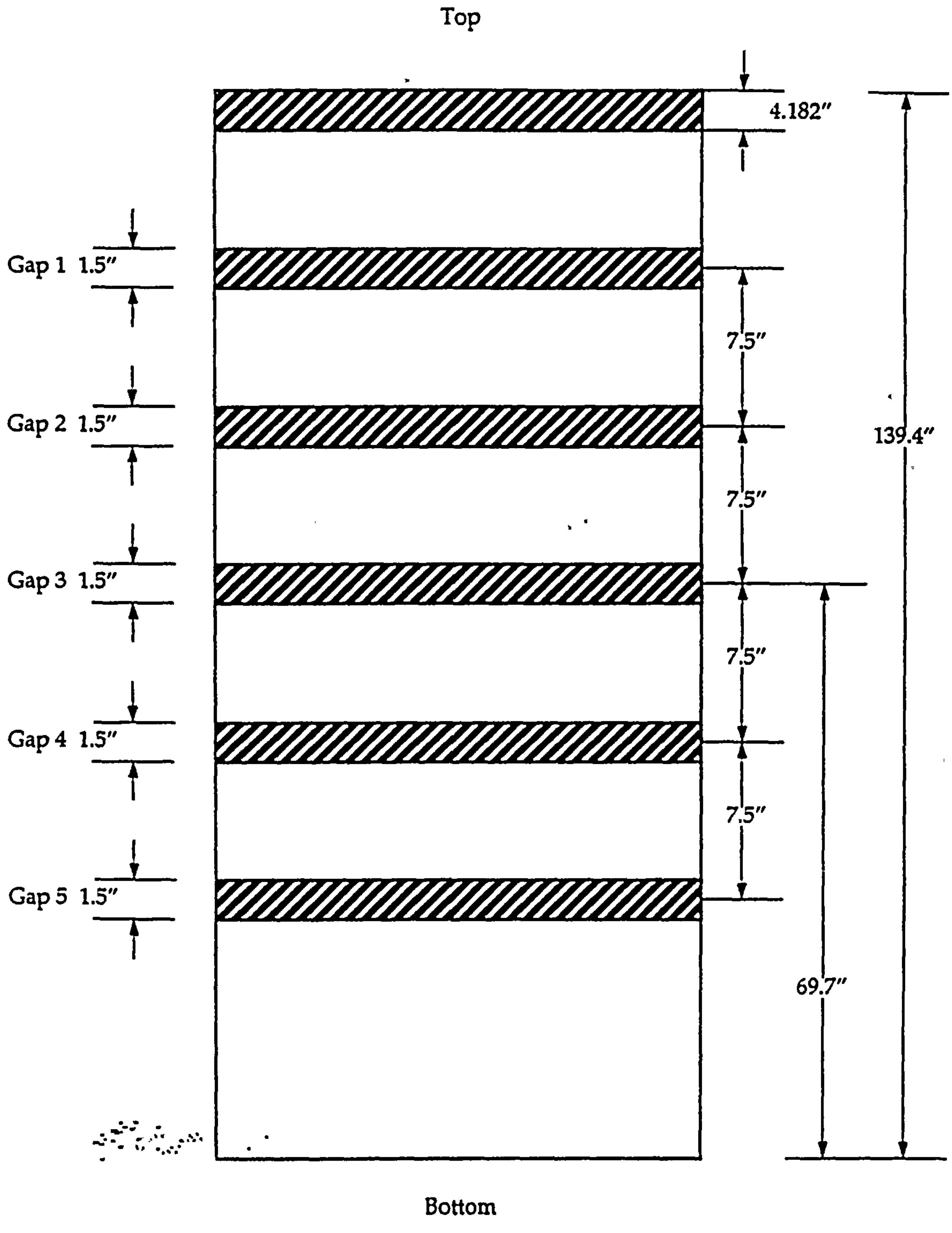
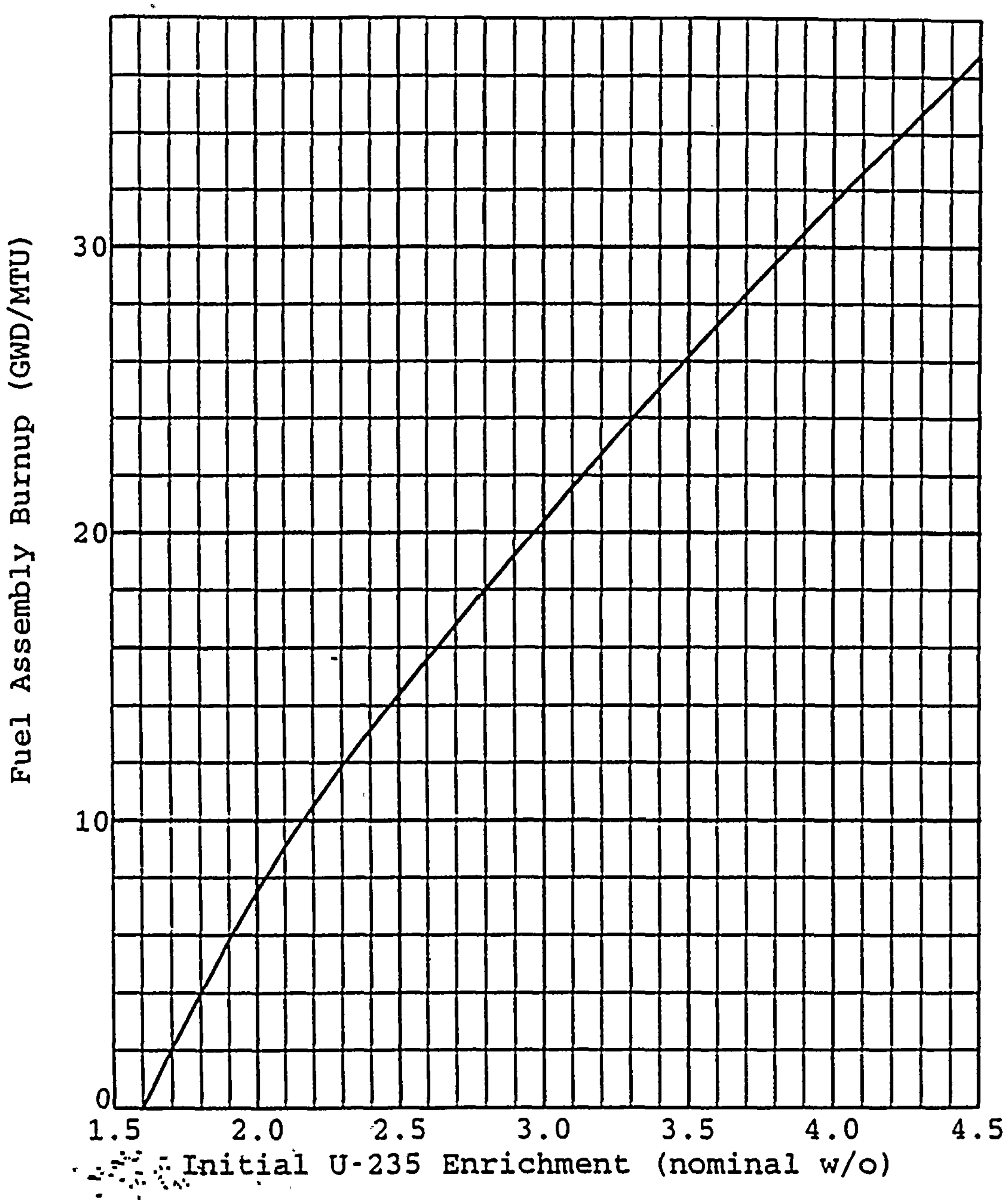


Figure 4. Turkey Point Units 3 and 4 High Density All Cell Configuration Burnup Credit Requirements (Region 2)



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ATTACHMENT 5

Criticality Analysis with Reduced B^{10} Loading in the Degraded Boraflex For Regions I and II Spent Fuel Storage

Criticality Analysis With a Reduced B^{10} Loading in the Degraded Boraflex for Turkey Point Units 3 & 4 Region 1 and Region 2 Spent Fuel All Cell Storage (No Soluble Boron)

October, 1999

S. Srinilta
S. Srinilta (ND)
Core Analysis B
Date: 10/5/99

Verified: JR Secker
J. Secker (ND)
Core Analysis C
Date: 10/5/99

Criticality Analysis With a Reduced B^{10} Loading in the Degraded Boraflex for Turkey Point Units 3 & 4 Region 1 and Region 2 Spent Fuel All Cell Storage (No Soluble Boron)

A criticality analysis was performed with a reduced B^{10} loading in the degraded boraflex for Turkey Point Units 3 & 4 Region 1 and Region 2 spent fuel all cell storage (No Soluble Boron). The methodology and assumptions used in the analysis are the same as in Reference 1 except that the absorber B^{10} loading and its thickness are reduced to 0.009 g/cm^2 and.0.0351 inch for Region 1. and 0.0006 g/cm^2 and 0.051 inch (remain unchanged) for Region 2. For Region 1, the reduction of both the B^{10} loading and the corresponding thickness is slightly more limiting than the reduction of the B^{10} loading only. For Region 2, the reduction of the B^{10} loading only is slightly more limiting than the reduction of the B^{10} loading and the corresponding thickness. The final 95/95 Keff is shown in the attached Table 1 and Table 2 for spent fuel rack Region 1 and Region 2, respectively. Since both Keff's are still less than 1.0, the Turkey Point Units 3 and 4 spent fuel racks will remain subcritical when all cells are loaded 15x15 fresh fuel assemblies with nominal enrichments no greater than 4.50 w/o U^{235} with natural uranium axial blankets in Region 1, and with nominal enrichments no greater than 1.60 w/o in Region 2. This meets the design basis for no soluble boron water in the pool.

Reference: 1) 99FP-G-0071 Criticality for Spent Fuel Storage for Turkey Point Units 3 & 4 (Degraded Boraflex)

Table 1. Region 1 - No Soluble Boron

Base Keno Reference Reactivity		0.97155

Calculation and Methodology Biases	**Range**	
Methodology (Benchmark) Bias		0.00770
Pool Temperature Bias	50 F to 185 F	0.00077
Boron Particles in Boraflex		0.00384
Total Bias	—	0.01231

Tolerances and Uncertainties	**Parameter Variation**	**Reactivity Variation**
Fuel Enrichment	+0.05/-0.05 %	0.00191
Fuel Density	+2/-2 %	0.00250
Fuel Pellet Dishing	-1.187 %	0.00145
Rack Cell Inner Dimension	+0.05/-0.025 inch	0.00153
Rack Cell Pitch	+0.12/-0.12 inch	0.01022
Rack Wall Thickness	+0.007/-0.007 inch	0.00024
Wrapper Plate Thickness	+0.002/-0.002 inch	0.00000
Poison Panel Thickness	+0.007/-0.007 inch	0.00973
Poison Cavity Thickness	+0.010/-0.010 inch	0.00004
Poison Panel Width	+0.075/-0.075 inch	0.00047
Asymmetric Assembly Position		0.00534
Calculation Uncertainty		0.00129
Benchmark Bias Uncertainty		0.00300
Total Uncertainty (convoluted)		0.01590

Final K_{eff} on 95/95 Basis	0.99976

Table 2. Region 2 - No Soluble Boron

Base Keno Reference Reactivity		0.97383
Calculation and Methodology Biases	**Range**	
Methodology (Benchmark) Bias		0.00770
Pool Temperature Bias	50 F to 185 F	0.00103
Boron Particles in Boraflex		0.00450
Total Bias		0.01323
Tolerances and Uncertainties	**Parameter Variation**	**Reactivity Variation**
Fuel Enrichment	+0.05/-0.05 %	0.00972
Fuel Density	+2/-2 %	0.00254
Fuel Pellet Dishing	-1.187 %	0.00116
Rack Cell Inner Dimension	+0.025/-0.025 inch	0.00000
Rack Cell Pitch	+0.07/-0.03 inch	0.00116
Rack Wall Thickness	+0.007/-0.007 inch	0.00000
Wrapper Plate Thickness	+0.002/-0.002 inch	0.00000
Poison Panel Thickness	+0.007/-0.007 inch	0.00582
Poison Cavity Thickness	+0.010/-0.010 inch	0.00000
Poison Panel Width	+0.075/-0.075 inch	0.00026
Asymmetric Assembly Position		0.00000
Calculation Uncertainty		0.00041
Benchmark Bias Uncertainty		0.00300
Total Uncertainty (convoluted)		0.01213
Final K_{eff} on 95/95 Basis		0.99919

ATTACHMENT 6

Turkey Point Units 3 and 4
Spent Fuel Pool Dilution Analysis

TURKEY POINT UNITS 3 AND 4

SPENT FUEL POOL DILUTION ANALYSIS

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Kenneth P. Slaby, Fluid Systems Engineering

Rev. 0

November 17, 1999

Table of Contents

1.0 INTRODUCTION

A boron dilution analysis has been completed for crediting boron in the Turkey Point Units 3 and 4 spent fuel rack criticality analysis. The boron dilution analysis includes an evaluation of the following plant specific features:

- Dilution Sources
- Boration Sources
- Instrumentation
- Administrative Procedures
- Piping
- Loss of Offsite Power Impact
- Boron Dilution Initiating Events
- Boron Dilution Times and Volumes

The boron dilution analysis was completed to ensure that sufficient time is available to detect and mitigate the dilution before the spent fuel rack criticality analysis 0.95 k_{eff} design basis is exceeded.

2.0 SPENT FUEL POOL AND RELATED SYSTEM FEATURES

This section provides background information on the spent fuel pool and its related systems and features. A one-line diagram of the spent fuel pool related systems is provided as Figure 1. For the purposes of this evaluation, the spent fuel pool and its related systems are sufficiently similar between the two Units that they will be treated as identical. Any significant differences will be identified, so that this report will be bounding for both Units.

2.1 Spent Fuel Pool

The design purpose of the spent fuel pool is to provide for the safe storage of irradiated fuel assemblies. The pool is filled with borated water. The water removes decay heat, provides shielding for personnel handling the fuel, and reduces the amount of radioactive gases released during a fuel handling accident. Pool water evaporation takes place on a continuous basis, requiring periodic makeup. The makeup source can be unborated water, since the evaporation process does not carry off the boron. Evaporation actually increases the boron concentration in the pool.

The spent fuel pool is a reinforced concrete structure with a minimum ¼" inch welded steel liner. The water-tight liner has dedicated drain lines to collect and detect liner leakage. The pool structure is designed to meet seismic requirements. The pool is approximately 38 feet deep. The top of the pit is located on the 58' elevation of the fuel handling building. The bottom of the pit is approximately at the 18' elevation.

In the event of excessive makeup flow into the pool, the pool would overflow onto the floor. There are no floor drains. Water would slowly drain into the transfer canal through metal covers which are not water-tight. The transfer canal is normally isolated from the SFP and empty. If the makeup rates exceeds the drain rate into the canal, the water level would rise approximately 3" to the bottom of the room door. Water would seep through the normally closed door to the outside roof of the fuel building. The open transfer canal and leakage through the room door minimize the effect of pool dilution sources, if any, from the floor elevation level.

As shown in Figure 2, the transfer canal lies adjacent to the pool and connects to the reactor refueling water cavity during refueling operations. The gates between the pool and the transfer canal are

normally closed. The volume of the pit is approximately 39,946 ft^3 to the Tech Spec minimum level elevation of 56'-10" less instrument accuracy. The majority of the water volume displaced by objects in the pit is by the spent fuel assemblies. The maximum number of assembly locations is 1404. The volume of all 1404 assemblies (3707 ft^3) is subtracted from the total pit volume. The racks themselves occupy a relatively small volume(403 ft^3), but they are subtracted as well. Finally, it is ascumed that a spent fuel cask is loaded into the pit, which displaces a small volume(429 ft^3). When the above volumes are subtracted from the pit volume, the remaining water volume (35;407 ft^3 = 264,845 gal.) is conservatively rounded down to 264,000 gallons.

2.2 Spent Fuel Storage Racks

The spent fuel racks are designed to support and protect the spent fuel assemblies under normal and credible accident conditions. Their design ensures the ability to withstand combinations of dead loads, live loads (fuel assemblies), and seismic loads.

2.3 Spent Fuel Pool Cooling System

The spent fuel pool cooling system is designed to remove the heat generated by stored spent fuel elements from the spent fuel pool. The system design incorporates redundancy for the only active component, the spent fuel pool cooling pump. System piping is configured so that failure of any pipeline in the cooling system does not drain the spent fuel pool below the top of the stored spent fuel assemblies.

The portion of the spent fuel pool cooling system which, if it failed, could result in a significant release of pool water is seismically designed.

The cooling system train consists of redundant pumps, a heat exchanger, valves, piping and instrumentation. The pump takes suction from the fuel pool at an inlet located below the pool water level, transfers the pool water through a heat exchanger and returns it back into the pool through an outlet located below and a large distance away from the cooling system inlet. The return line is designed to prevent siphoning. The heat exchangers are cooled by component cooling water.

2.4 Spent Fuel Pool Cleanup System

The spent fuel pool cleanup system is designed to maintain water clarity and to control borated water chemistry. The cleanup system is connected to the spent fuel pool cooling system. About 100 gpm of the spent fuel pool cooling pump(s) discharge flow can be diverted to the cleanup loop, which includes the spent fuel pool demineralizer and filters. The filters remove particulates from the spent fuel pool water and the spent fuel pool demineralizer removes ionic impurities.

The refueling water purification loop also uses the spent fuel pool demineralizer and filters to clean up the refueling water storage tank after refueling operations. The design flow rate in the loop is limited to 100 gpm to accommodate the design flow of the spent fuel pool demineralizer.

The spent fuel pool has a surface skimmer system designed to provide optical clarity by removing surface debris. The system consists of two surface skimmers, a single strainer, a single pump and three filters. The skimmer pump is a centrifugal pump with a 100 gpm capacity. The pump discharge flow passes through the filter to remove particulates. It returns to the spent fuel pool.

2.5 Dilution Sources

2.5.1 Chemical and Volume Control System (CVCS)

The CVCS connects to the spent fuel pool via a 2" line from the discharge of the outlet of the holdup tanks recirculation pump to the refueling water purification pump bypass line to the spent fuel pool demineralizer inlet and into the cooling loop return header. This connection is normally isolated and is used to transfer water from the holdup tanks to the spent fuel pool. The isolation is by three manual valves.

Since there is also a check valve at the recirculation pump discharge, water will not flow from the spent fuel pool to the holdup tanks. Also, holdup tank water will not gravity-drain to the spent fuel pool because the holdup tank recirculation pump is normally isolated, and the maximum tank water level is below the minimum SFP level.

The recirculation pump can take suction from either of the three holdup tanks. However, by procedure, the only one pump is aligned to one holdup tank at a time. Manual valve manipulations are required to switch the pump suction to another tank. Each holdup tank has a total volume of approximately 97,000 gallons and can be at a boron concentration from 0 ppm up to 2000 ppm. The flow from this source is estimated to be 90 gpm.

2.5.2 Primary Water Makeup System

The primary water makeup system consists of one primary water storage tank and two primary water pumps per Unit. During normal operation, one primary water pump is running on recirculation to provide primary water on demand to multiple users. Each primary water storage tank contains approximately 150,000 gallons of non-borated, demineralized water.

The primary water makeup system connects to the spent fuel pool directly via the cooling loop return line, and indirectly through the spent fuel pool demineralizer outlet and the local station in the spent fuel pit area. Using the direct connection, the contents of the primary water storage tank can be transferred directly to the spent fuel pool cooling system via the primary water pumps. The direct connection is normally isolated from the primary water system by three locked-closed manual valves. The preferred method of makeup is from the refueling water storage tank which is borated. The second preferred makeup method is from the direct connection to the spent fuel pool cooling loop return header. The flow rate through this path is estimated to be 415 gpm.

When primary water is used to flush spent resin, the spent fuel pool demineralizer is isolated from the cleanup loop by one manual valve. If this valve were left open, primary water could be transferred into the spent fuel pool. The flow from this pathway is estimated to be 240 gpm.

Finally, the 2" primary water station in the spent fuel pit area is isolated by a normally closed valve and a capped connection. The flow from this pathway is estimated to be 500 gpm.

2.5.3 Demineralized Water System

The demineralized water system is supplied from a water treatment plant. This source of makeup is utilized only when the treatment plant supply pressure is at least 50 psig. Demineralized water is

provided through the same connection to the spent fuel pool cooling loop return piping as the primary water source. The demineralized water supply is isolated by one normally closed and one locked closed valve. The flow from this source is estimated to be 174 gpm.

2.5.4 Component Cooling Water System

Component cooling water is the cooling medium for the spent fuel pool cooling system heat exchanger. There is no direct connection between the component cooling system and the spent fuel pool cooling system. If, however, a leak were to develop in a heat exchanger that is in service, the connection would be made. Since the component cooling system normally operates at a slightly higher pressure than the spent fuel pool cooling system, it is expected that a breach in a spent fuel pool cooling system heat exchanger tube would result in non-borated component cooling water entering the spent fuel pool cooling system.

It would be expected that the flow rate of any leakage of component cooling water into the spent fuel pool cooling system would be very low due to the small difference in operating pressures between the two systems. Even if there was significant leakage from the component cooling water system to the spent fuel pool, the impact on the spent fuel pool boron concentration would be minimal because a loss of water from the component cooling water surge tank would initiate an alarm and control room indication to alert the control room operators.

If the alarms which would alert the control room operators of a component cooling water system leak were to fail and leakage from the component cooling water system to the spent fuel pool cooling system were to continue undetected, the component cooling water surge tank would be administratively refilled with primary water. Until makeup is initiated, the volume added to the spent fuel pool would be limited by the component cooling water surge tank volume of 2000 gallons.

Because of the limited dilution volume from this source relative to the spent fuel pool volume, it is not considered further in this analysis.

2.5.5 Drain Systems

The equipment drain system connects directly to the spent fuel pool cooling system and skimmer system at the drain connections for the spent fuel pit pumps, heat exchangers (tube side), filters, demineralizer, the skimmer pump, and skimmer filter. Each connection has a normally closed valve to isolate it. Backflow through these paths is not considered credible, because if the drain valves were left open, the pressurized spent fuel pool cooling system would flow into the drain system, not vice versa.

2.5.6 Fire Protection System

In an emergency loss of spent fuel pool inventory, a fire hose station is available outside the spent fuel pool area door. This station is capable of providing 100 gpm of non-borated water. Although an available source, the fire hose is not specifically addressed by an approved procedure for makeup.

2.5.7 Spent Fuel Pit Demineralizer

The spent fuel pit demineralizer has a capacity of 30 ft^3 of 1:1 equivalent mixed bed resin. This implies a volume ratio of 60%/40% anion to cation resin. If we assume the bed was loaded with 100% anion, it would bound the capacity to remove boron when it is first aligned to the system. The demineralizer would be operated at a nominal 100 gpm flow rate. Dilution of the spent fuel pool resulting from operation of the demineralizer will not result in an increase in the spent fuel pool level.

2.5.8 Rainwater Collection System

The room housing the spent fuel pool includes 4-6" piping along the walls to carry rainwater from the roof to a drain system. The piping is not seismically designed and could therefore be postulated to break during an earthquake.

2.5.9 Dilution Source and Flow Rate Summary

Based on the evaluation of potential spent fuel pool dilution sources summarized above, the following dilution sources were determined to be capable of providing a significant amount of non-borated water to the spent fuel pool. The potential for these sources to dilute the spent fuel pool boron concentration to the design basis boron concentration (650 ppm) will be evaluated in Section 3.0.

SOURCE	APPROXIMATE FLOW RATE (GPM)
CVCS	
- Holdup Tank to Cleanup Loop	90
Primary Water System	
- To SFP via valve 821	415
- To SFP via demineralizer sluice line	240
- 2" PW station near SFP	500
Demineralized Water System	
- To SFP via valve 821	174
Fire Protection System	
- Fire hose station outside SFP room	100
SFP Demineralizer	100

2.6 Boration Sources

The normal source of borated water to the spent fuel pool is from the refueling water storage tank. It is also possible to borate the spent fuel pool by the addition of dry boric acid directly to the spent fuel pool water.

2.6.1 Refueling Water Storage Tank

The refueling water storage tank (RWST) connects to the spent fuel pool through the purification loop via the refueling water purification pump. This connection is used to purify the RWST water when the purification loop is isolated from the spent fuel pool cooling system. Normally, this connection can supply borated water to the spent fuel pool via the refueling water purification pump to the inlet to the spent fuel pit cooling system purification loop. The refueling water purification pump is powered from a non-vital bus power supply. The RWST is required by Technical Specifications to be kept at a minimum boron concentration of 1950 ppm.

2.6.2 Direct Addition of Boric Acid

If necessary, the boron concentration of the spent fuel pool can be increased by emptying bags of dry boric acid directly into the spent fuel pool. However, boric acid dissolves very slowly at room temperature and requires that the spent fuel pool cooling pumps be available for mixing the spent fuel pool water (see section 3.1 for further discussion on spent fuel pool mixing.) Furthermore, there is no procedure currently in place to provide operator guidance for this method. Therefore, this method would be used only in an emergency.

2.7 Spent Fuel Pool Instrumentation

Instrumentation is available to monitor spent fuel pool water level and temperature. Additional instrumentation is provided to monitor the pressure and flow of the spent fuel pool cleanup system, and pressure, flow, and temperature of the spent fuel pool cooling system.

The instrumentation provided to monitor the temperature of the water in the spent fuel pool is indicated locally and annunciated in the control room. The water level instrumentation alarms, high and low level, are annunciated in the control room. Two area radiation monitors are available in the spent fuel pool room.

A change of one foot in spent fuel pool level with the transfer canal isolated requires approximately 7816 gallons of water. If the pool level was raised from the low level alarm point to the high level alarm (6" including instrument error), a dilution of approximately 3908 gallons could occur before an alarm would be received in the control room. If the spent fuel pool boron concentration were at 1950 ppm initially, such a dilution would only result in a reduction of the pool boron concentration of approximately 28 ppm.

2.8 Administrative Controls

The following administrative controls are in place to control the spent fuel pool boron concentration and water inventory:

1. Procedures are available to aid in the identification and termination of dilution events.

2. The procedures for loss of inventory (other than evaporation) specify that a borated makeup

source (RWST) be used as the preferred makeup source. The procedures specify non-borated sources as secondary preferences.

3. In accordance with procedures, plant personnel perform rounds in the spent fuel pool room once every 12 hours. The personnel making rounds to the spent fuel pool are trained to be aware of the change in the status of the spent fuel pool. They are instructed to check the temperature and level in the pool and conditions around the pool during plant rounds.

4. Administrative controls (locked closed valves on primary water and demineralized water flow paths to the spent fuel pool cooling system) are placed on some of the potential dilution paths.

5. Procedures require that the chemistry department verify the pool boron-concentration will not be diluted below 1950 ppm when makeup is added.

2.9 Piping

There are no systems (other than those listed in section 2:5.1 to 2.5.8) identified which have piping in the vicinity of the spent fuel pool which could result in a dilution of the spent fuel pool if they were to fail.

2.10 Loss of Offsite Power Impact

Of the dilution sources listed in Section 2.5.9, only the fire protection system is capable of providing non-borated water to the spent fuel pool during a loss of offsite power.

The loss of offsite power would affect the ability to respond to a dilution event. The spent fuel pool level instrumentation is not powered from vital power supplies.

The refueling water purification pump is not powered from a safeguards supply and would not be available to deliver borated water from the RWST. However, at maximum water level, the RWST can be gravity-drained to the spent fuel pool through the refueling water purification pumps, if necessary, to provide a borated water source. Finally, manual addition of dry boric acid to the pool could be used if it became necessary to increase the spent fuel pool boron concentration during a loss of offsite power.

3.0 SPENT FUEL POOL DILUTION EVALUATION

3.1 Calculation of Boron Dilution Times and Volumes

For the purposes of evaluating spent fuel pool dilution times and volumes, the total pool volume available for dilution, as described in section 2.1, is conservatively assumed to be 264,000 gallons.

Based on the criticality analyses (Reference 1), the soluble boron concentration required to maintain the spent fuel pool boron concentration at $K_{eff} < 0.95$, including uncertainties and burnup, with a 95% probability at a 95% confidence level (95/95) is 650 ppm.

The spent fuel pool boron concentration is typically in the range of 2000 and 2100 ppm. If the concentration falls below 1950 ppm, Turkey Point enters a Limiting Condition of Operation Action Requirement procedure and uses administrative procedures to restore and monitor the concentration. However, for the purposes of evaluating the dilution times and volumes, the initial spent fuel pool boron concentration is assumed to be at the current Technical Specification minimum limit of 1950 ppm. The evaluations are based on the spent fuel pool boron concentration being diluted from 1950 ppm to 650 ppm. To dilute the combined pool volume of 264,000 gallons from 1950 ppm to 650 ppm would conservatively require 290,000 gallons of non-borated water, based on a feed-and-bleed operation (constant volume).

This analysis assumes thorough mixing of all the non-borated water added to the spent fuel pool with the contents of the spent fuel pool. Refer to Figure 3. Based on the design flow of 2300 gpm per spent fuel pit pump, the 264,000 gallon system volume is turned over approximately every two hours with one pump running, which is the normal alignment. It is unlikely, with cooling flow and convection from the spent fuel decay heat, that thorough mixing would not occur. However, if mixing was not adequate, it would be conceivable that a localized pocket of non-borated water could form somewhere in the spent fuel pool. This possibility is addressed by the calculation in Reference 1 which shows that the spent fuel rack K_{eff} will be less than 1.0 on a 95/95 basis with the spent fuel pool filled with non-borated water. Thus, even if a pocket of non-borated water formed in the spent fuel pool, K_{eff} would not exceed 1.0 anywhere in the pool.

The time to dilute the spent fuel pool depends on the initial volume of the pool and the postulated rate of dilution. The dilution volumes and times for the dilution scenarios discussed in Sections 3.2 and 3.3 are calculated based on the following equation:

$$t_{end} = \ln (C_o / C_{end}) V/Q \qquad \text{(Equation 1)}$$

Where:

C_o = the boron concentration of the pool volume at the beginning of the event (1950 ppm)

C_{end} = the boron endpoint concentration (650 ppm)

Q = dilution rate (gallons/minute)

V = volume (gallons) of spent fuel pool (264,000)

t_{end} = time to reach C_{end} (minutes)

3.2 Evaluation of Boron Dilution Events

The potential spent fuel pool dilution events that could occur are evaluated below:

3.2.1 Dilution From CVCS Holdup Tanks

The contents of a CVCS holdup tank can be transferred via the recirculation pump to the spent fuel pool via the cleanup loop. The flow path to the transfer canal is through a line that is isolated by one normally closed valve. This connection is a designated source of makeup water in a loss of spent fuel pool inventory event. Each of the three CVCS recycle holdup tanks has a total volume of approximately 97,000 gallons. The water in the tanks can have a boron concentration from 0 ppm to approximately 2000 ppm. Any amount of boron in the CVCS holdup tank water would reduce the dilution of the spent fuel pool resulting from the transfer of CVCS holdup tank water to the spent fuel pool. To dilute the spent fuel pool volume from 1950 ppm to 650 ppm would require 290,000 gallons of unborated water. The combined contents of the three CVCS holdup tanks (approximately 300,000 gallons) is slightly more than the required dilution volume. The path from the recirculation pump to the spent fuel pool via the connection to the spent fuel pool purification loop can provide approximately 90 gpm. If the manual isolation valve were left unattended, it would take 43 minutes to increase the spent fuel pool level from the low to high alarm setpoints, and 54 hours to provide the 290,000 gallons required to dilute the pool from 1950 to 650 ppm boron, assuming 0 ppm boron in the holdup tanks.

The CVCS recirculation pump can take suction from either of the three CVCS holdup tanks. Administrative procedures specify that the pumps are aligned to one holdup tank at a time. Manual valve manipulations are required to switch the pump suction to another tank. Thus, it is assumed for the purposes of this evaluation that only the contents of one CVCS holdup tank is available for a spent fuel pool dilution event. The 97,000 gallons of water contained in one CVCS holdup tank is less than the 290,000 gallons necessary to dilute the spent fuel pool/transfer canal from 1950 ppm to 650 ppm. Because of these factors, the CVCS holdup tanks are not considered a credible dilution source for the purposes of this analysis.

3.2.2 Dilution From Primary Water Storage Tanks

The contents of the primary water storage tank can be transferred via the primary water pumps directly or indirectly to the spent fuel pool.

The primary water system consists of a primary water storage tank and two primary water pumps per Unit. Primary water can be supplied to the spent fuel pool cooling system from the tank and pumps associated with either Unit. The two primary water storage tanks each contain approximately 150,000 gallons of non-borated reactor grade water. The tanks are normally not cross-connected. Thus, the contents of one tank is not sufficient to dilute the spent fuel pool from 1950 to 650 ppm.

The path from the primary water pumps to the spent fuel pool via the connection to the spent fuel pool cooling loop return header can provide approximately 415 gpm. If the manual isolation valve were left unattended, it would take 9 minutes to increase the spent fuel pool level from the low to high alarm setpoints, and 12 hours to provide the 290,000 gallons required to dilute the pool from 1950 to 650 ppm boron.

The path from the primary water pumps to the spent fuel pool via the spent fuel pit demineralizer resin flushing connection can provide approximately 240 gpm. If the manual isolation valve were left unattended, it would take 16 minutes to increase the spent fuel pool level from the low to high alarm setpoints, and 20 hours to provide the 290,000 gallons required to dilute the pool from 1950 to 650 ppm boron.

The path from the primary water pumps to the spent fuel pool via the 2" station in the spent fuel pit area can provide approximately 500 gpm. If the temporary hose connection were left unattended, it would take 8 minutes to increase the spent fuel pool level from the low to high alarm setpoints, and 10 hours to provide the 290,000 gallons required to dilute the pool from 1950 to 650 ppm boron.

3.2.3 Dilution From Demineralized Water System

This source consists of a trailer-type portable system which is administratively controlled. Thus, its capacity and surge volume are very limited.

The path from the demineralized water connection to the spent fuel pool via the 2" connection to the spent fuel pool cooling system return header can provide approximately 174 gpm. If the path were left unattended, it would take 22 minutes to increase the spent fuel pool level from the low to high alarm setpoints, and 28 hours to provide the 290,000 gallons required to dilute the pool from 1950 to 650 ppm boron.

3.2.4 Dilution from Fire Protection System

The fire protection system draws from two raw water tanks with a capacity of 500,000 and 750,000 gallons, respectively. The path from the fire water pump to the spent fuel pool via the fire hose station outside the spent fuel pit area can provide approximately 100 gpm. If the hose were left unattended, it would take 39 minutes to increase the spent fuel pool level from the low to high alarm setpoints, and 48 hours to provide the 290,000 gallons required to dilute the pool from 1950 to 650 ppm boron.

3.2.5 Dilution Resulting From Seismic Events or Random Pipe Breaks

A seismic event could cause piping ruptures in the vicinity of the spent fuel pool in piping that is not seismically qualified. The only piping within the immediate vicinity of the spent fuel pool that could result in dilution of the spent fuel pool if it ruptures during a seismic event are the 4-6" rain collection piping along the walls. In order to consider this piping as a dilution source, it would be necessary to assume that a seismic event occurred coincident with a major rainstorm that would dump significant volumes of unborated rainwater into the spent fuel pool. This combination of events is considered to have a sufficiently low probability of occurrence to eliminate it from consideration for this analysis.

For a seismic event with offsite power available, rupture of the primary water station piping in the spent fuel pit area are bounded by the analyses in Sections 3.2.2. If offsite power is not available, the primary water systems would not operate, and thus, there would be no dilution source.

3.2.6 Dilution From Spent Fuel Pool Demineralizer

When the spent fuel pool demineralizer is first placed in service after being recharged with fresh resin, it can initially remove boron from the water passing through it. In the worst case, assuming 30 ft^3 of anion resin in the demineralizer, up to 18 ppm of boron could be removed from the spent fuel pool water before the resin would become saturated. Since the demineralizer normally utilizes a mixed bed of anion and cation resin, less boron would actually be removed before saturation. Because of the small amount of boron removed by the demineralizers, it is not considered a credible dilution source for the purposes of this evaluation.

3.2.7 Review of Licensee Event Reports(LER)

A review of 8 LERs related to the spent fuel pool and cooling system was conducted to identify any extraordinary mechanisms of dilution not previously addressed in the previous sections. The review resulted in no new dilution paths or mechanisms.

3.3 Summary of Dilution Events

The four available water sources for spent fuel pool dilution are primary water, demineralized water, CVCS holdup tank fluid, and fire protection. Fire protection is the least likely source, since it is not controlled by procedure for use as makeup, and because the fire hose is located outside the spent fuel pool room. The CVCS holdup tank source is the next least likely because it is normally borated to some degree, and because the volume of one tank is less than that required to dilute the spent fuel pool from 1950 to 650 ppm. The demineralized water source is the next least likely source because its source is controlled administratively, and because it is third in the preferred list of normal makeup sources behind the RWST and the primary water system. Since the RWST is always borated, the key system for consideration is the primary water system.

Flow rates from the primary water system supply pump vary up to pump runout flow of 500 gpm. Even at this flow, the spent fuel pool is filled to the high alarm in only 8 minutes, assuming the pool level was initially at the Technical Specification minimum level. Assuming that the high level alarm were to fail, the pool would overflow and fill the room until water would leak into the refueling canal and out of the room door leaking to the building roof. A significant loss of primary water storage tank inventory would be detected by the operator via a low tank level alarm at 50%. Makeup to the tank is provided administratively from the deaeration system, so the tank would not be refilled continuously without operator attention. Finally, the total tank volume of 150,000 gallons is not sufficient to dilution the spent fuel pool from 1950 to 650 ppm boron.

Furthermore, for any dilution scenario to successfully add 290,000 gallons of water to the spent fuel pool, plant operators would have to fail to question or investigate the continuous makeup of water to the primary water storage tank for the required time period, and fail to recognize that the need for 290,000 gallons of makeup was unusual.

4.0 CONCLUSIONS

A boron dilution analysis has been completed for the spent fuel pool. As a result of this spent fuel pool boron dilution analysis, it is concluded that an unplanned or inadvertent event which would result in the dilution of the spent fuel pool boron concentration from 1950 ppm to 650 ppm is not a credible event. This conclusion is based on the following:

The preferred method of normal makeup to the pool is, by procedure, from the RWST, a borated water source. Thus, the operator would have to make a conscious decision to choose a non-borated water source for makeup over a borated water source.

If an inadvertent dilution were to be initiated, administrative procedures are in place to address a high level alarm in the spent fuel pool. Borated water from the RWST is available via the refueling water purification pump with normal power available, and by gravity feed to the pool should offsite power be lost.

In order to dilute the spent fuel pool to the design k_{eff} of 0.95, a substantial amount of water (290,000 gallons) is needed. To provide this volume, an operator would have to initiate the dilution flow, then abandon monitoring of pool level, and ignore administrative procedures, and a high level alarm for a period of at least 10 hours. The 290,000 gallon value is used for analysis. It exceeds the volume of the typical source of unborated water (primary water storage tank). There is no single credible source of unborated water of at least 290,000 gallons.

Since such a large water volume turnover is required, a spent fuel pool dilution event would be readily detected by plant personnel via alarms, flooding in the fuel handling building, or eventually by operator rounds through the spent fuel pool area.

It should be noted that this boron dilution evaluation was conducted by evaluating the time and water volumes required to dilute the spent fuel pool from 1950 ppm to 650 ppm. The 650 ppm endpoint was utilized to ensure that K_{eff} for the spent fuel racks would remain less than or equal to 0.95. As part of the criticality analysis for the spent fuel racks (Reference 1), a calculation has been performed on a 95/95 basis to show that the spent fuel rack K_{eff} remains less than 1.0 with non-borated water in the pool. Thus, even if the spent fuel pool were diluted to zero ppm, which would take significantly more

water than evaluated above, the spent fuel would be expected to remain subcritical and the health and safety of the public would be assured.

5.0 REFERENCES

1. CAB-99-214, "Criticality for Fresh and Spent Fuel Storage for Turkey Point Units 3 and 4 (Full Boraflex)," June, 1999.

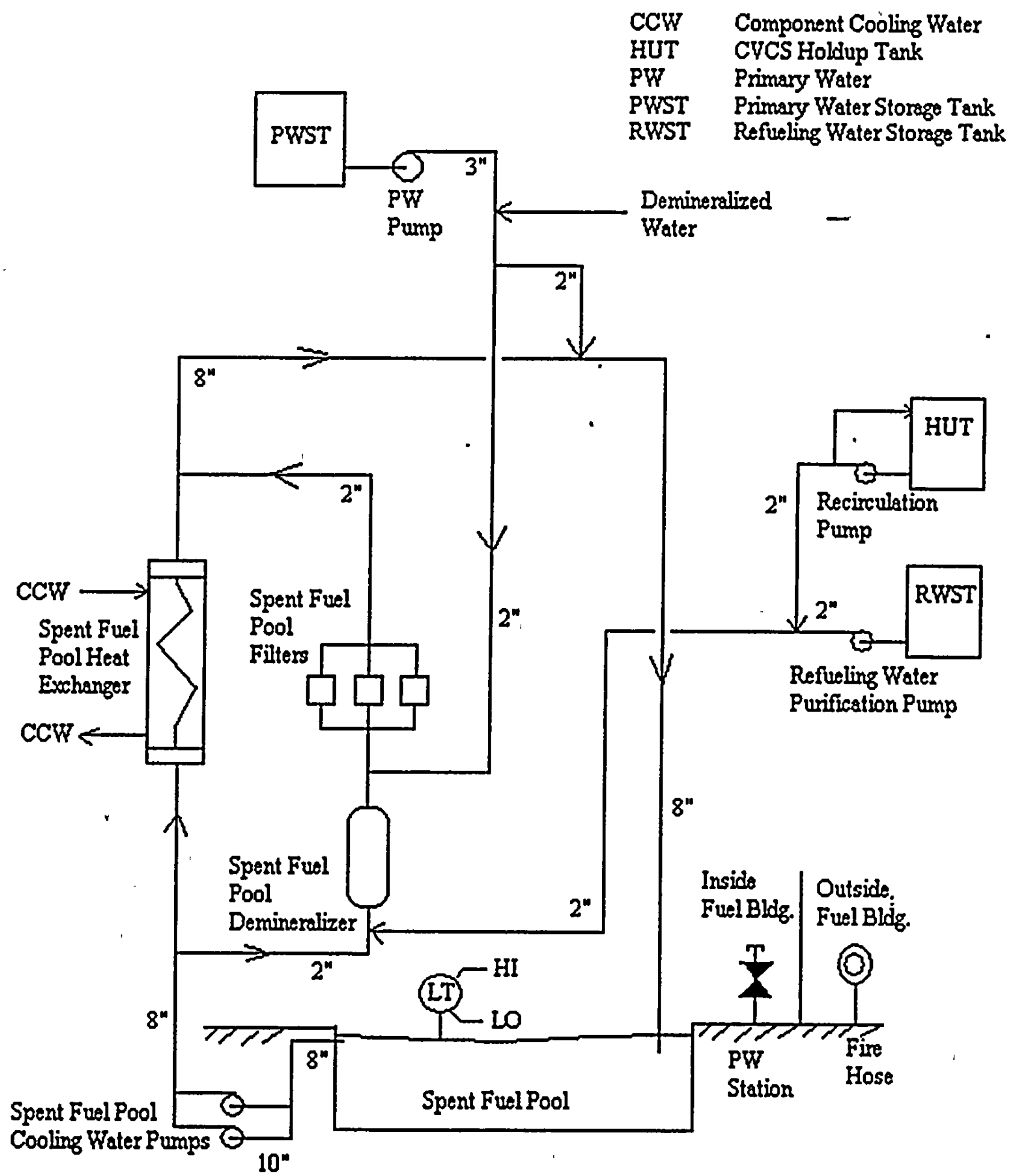

Figure 1 - Spent Fuel Pool and Related Systems

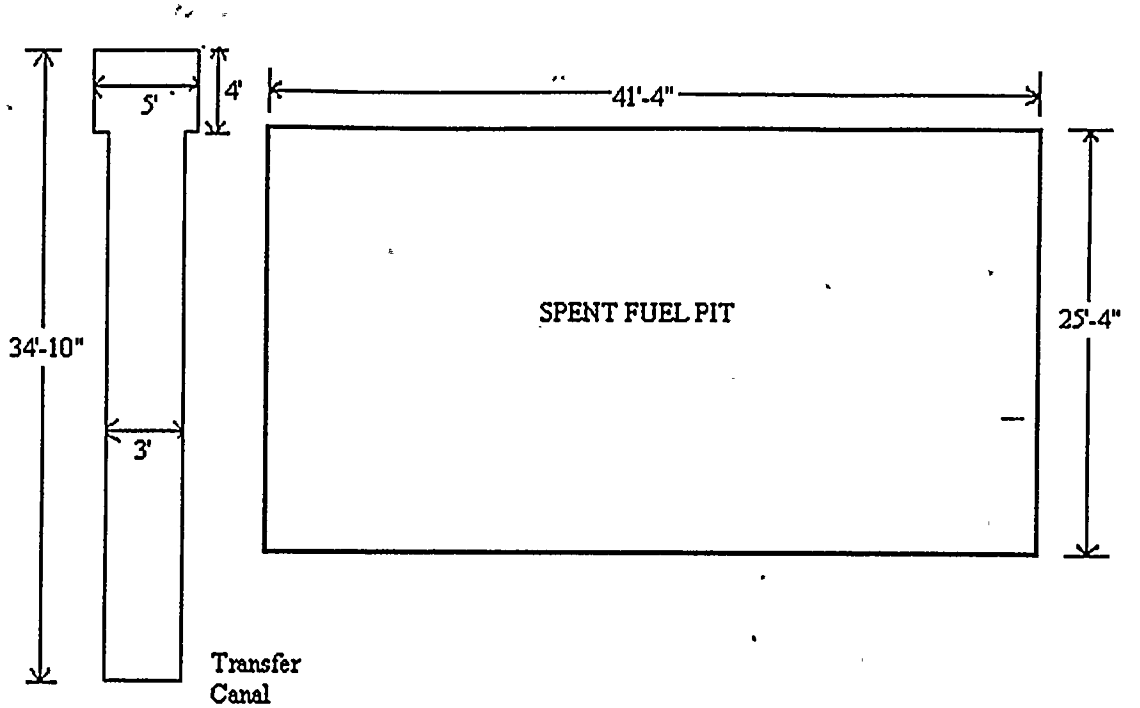

Figure 2 - SFP Plan View 1

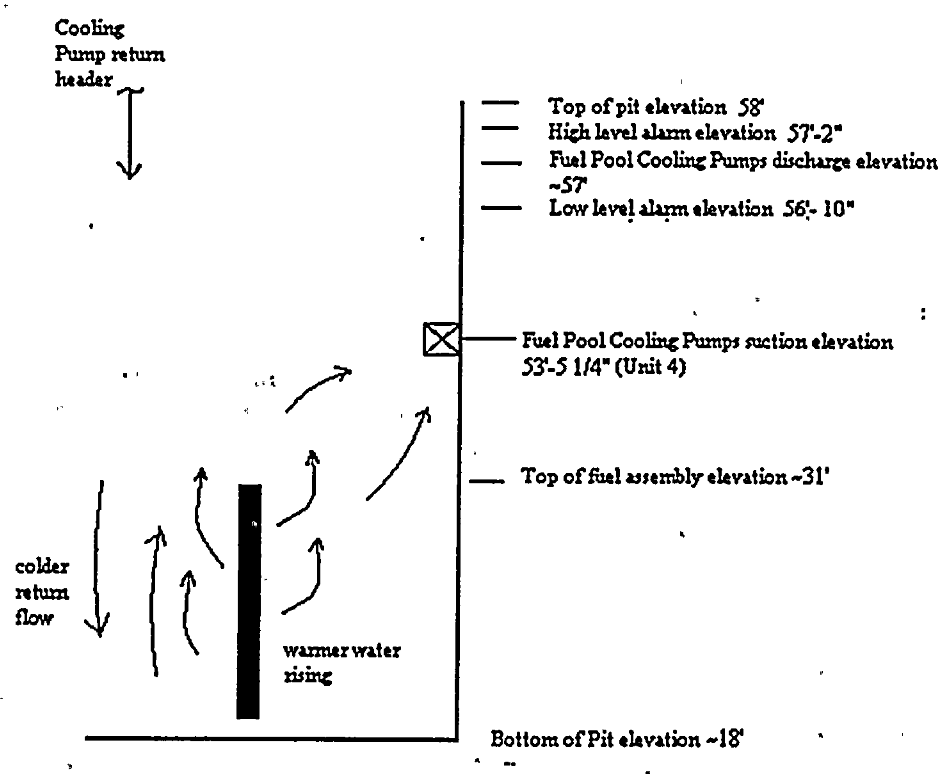

Figure 3 - SFP Mixing 1

ATTACHMENT 7

Criticality for Fresh Fuel Storage for Turkey Point Units 3 & 4

CAB-99-426

Criticality for Fresh Fuel Storage for Turkey Point Units 3 & 4

October 1999

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1.0 Introduction

This report presents the results of criticality analyses of the Florida Power & Light Turkey Point Units 3 and 4 fresh fuel storage vault.

The following storage configuration and enrichment limit were considered in the analyses:

Fresh Fuel	Storage of 15x15 fuel assemblies in the fresh fuel vault. Fuel assemblies must have a nominal enrichment no greater than 4.5 w/o ^{235}U with no axial blankets.

Under full density water flooding accident, the storage is shown to meet the limit of $K_{eff} \leq 0.95$ and for the optimum moderation accident the limit of $K_{eff} \leq 0.98$ is met.

1.1 Design Description

The cross-sectional view of the Turkey Point Unit 4 fresh fuel rack layout is depicted in Figure 1 on page 13. Fresh fuel storage cell layout for Turkey Point Unit 3 is a mirror image of Turkey Point Unit 4.

The fuel parameters relevant to this analysis are given in Table 1 on page 11 and Table 2 on page 12. The Table 1 fuel parameters bound the previous fuel designs used in Turkey Point Units 3 & 4. The fuel rod and guide thimble and instrumentation thimble tube cladding are modeled as zircaloy in this analysis. This is conservative with respect to the Westinghouse ZIRLO™ product which is a zirconium alloy containing additional elements including niobium. Niobium has a small absorption cross section which causes more neutron capture in the cladding resulting in a lower reactivity. Therefore, this analysis is conservative with respect to fuel assemblies containing ZIRLO™ cladding in fuel rods and guide thimble and instrumentation thimble tubes.

Table 2 on page 12 also shows the rack dimensions with the manufacturing tolerances.

The analyses are performed for full density water flooding and with a low density water presence in the fuel storage area which maximizes the storage reactivity.

The analyses conform to the requirements of NUREG-0800, Section 9.

1.2 Design Criteria

Criticality of fuel assemblies in a fuel storage rack is prevented by the design of the rack which limits fuel assembly interaction. This is done by fixing the minimum separation between fuel assembly and/or placing absorber panels between storage cells.

The design basis for preventing criticality for fresh fuel storage, is examined on the 95/95 basis. The 95/95 basis is defined as the upper limit, with a 95 percent probability at a 95 percent confidence level, of the effective neutron multiplication factor K_{eff} of the fuel assembly array, including uncertainties and manufacturing tolerances.

Fresh fuel is normally stored in dry conditions. However, the accidental introduction of water in the storage area is considered. For the full density water flooding in the fresh fuel storage area, the criterion of 95/95 basis $K_{eff} \leq 0.95$ must be met. For optimum moderation (water content which gives the highest reactivity) in the storage area the criterion of 95/95 basis $K_{eff} \leq 0.98$ must be met.

The design criteria are consistent with ANSI 57.2-1983[2], ANSI 57.3-1983[3], and NUREG-0800[4].

2.0 Analytical Methods

The criticality calculation method and cross-section values are verified by comparison with critical experiment data for fuel assemblies similar to those for which the racks are designed. This benchmarking data is sufficiently diverse to establish that the method bias and uncertainty will apply to rack conditions which include strong neutron absorbers, large water gaps, low moderator densities and spent fuel pool soluble boron.

The design method which insures the criticality safety of fuel assemblies in the fuel storage rack is described in detail in the Westinghouse Spent Fuel Rack Criticality Analysis Methodology topical report[1]. This report describes the computer codes, benchmarking, and methodology which are used to calculate the criticality safety limits presented in this report for Turkey Point Units 3 and 4.

As determined in the benchmarking in the topical report, the method bias using the described methodology of NITAWL-II, XSDRNPM-S and KENO-Va is 0.00770 ΔK with a 95 percent probability at a 95 percent confidence level uncertainty on the bias of 0.00300 ΔK. These values will be used in this report.

3.0 Criticality Analysis of the Fresh Fuel Racks

The fresh fuel is stored in the fresh fuel racks in a dry condition. The reactivity of the dry fuel is very low at enrichments up to 5 w/o. However, with any introduction of water in the storage area the reactivity of the array can rise significantly. Thus, the reactivity of the system is examined for the conditions of flooding and for introduction of water at optimum density. The optimum density is defined as the low density of water which would lead to the highest reactivity of the storage array.

This section describes the analytical techniques and models employed to perform the criticality analysis for the storage of fresh fuel in the Turkey Point Units 3 and 4 fresh fuel storage racks. The fresh fuel rack is analyzed by employing the methodology outlined in Section 2 of this report. Details of this analysis are outlined in Sections 3.1 and 3.2 for the full density water flooding condition and optimum moderation condition, respectively.

Fresh fuel is stored in racks arranged in an L-shaped configuration, consisting of two limbs of 8x3 and 3x10. Up to 54 assemblies can be stored. Figure 1 on page 13 shows the configuration.

Since the fresh fuel racks are normally maintained in a dry condition, the criticality analysis will show that the rack 95/95 K_{eff} is less than or equal to 0.95 for the accidental full density water flooding scenario (ANSI/ANS-57.3) and less than or equal to 0.98 for the accidental low water density (optimum moderation) flooding scenario (NUREG-0800, Section 9).

The analyses are performed for the fresh fuel storage in the "worst case" conditions. In these conditions, the manufacturing tolerances are included in the calculation input in the limiting direction, so that the calculations are conservative.

3.1 Full Density Moderation Analysis

The fresh fuel racks are analyzed for the full density water flooding condition under "worst case" scenario:

1. Maximum enrichment of 4.55 w/o ^{235}U
2. No pellet dishing
3. Fuel with 98% of theoretical density
4. Minimum assembly storage pitch of 20.875 inches.
5. An infinite array of assemblies (no radial leakage of neutrons) is modeled.
6. Top and bottom of the storage have 1 foot of water.

The "worst case" scenarios conservatively account for fuel parameter variability and tolerances on rack dimensions. The KENO results for the "worst case" model are then used to develop the maximum 95/95 K_{eff} which is compared to the criticality safety limit of 0.95.

Therefore, following assumptions were used to develop the KENO model for the storage of fresh fuel in the Turkey Point Units 3 and 4 fresh fuel storage racks under full density water condition:

1. The fuel assembly parameters for the criticality analysis are based on the Westinghouse 15x15 DRFA design.
2. All fuel rods contain uranium dioxide at the maximum enrichment of 4.55 w/o over the entire length of each rod.
3. The fuel pellets are modeled assuming a UO_2 density which is 98% of theoretical density with no dishing fraction (0%) for "worst case" conditions.
4. No credit is taken for any ^{234}U or ^{236}U in the fuel.
5. No credit is taken for any grids in the assembly.
6. No credit is taken for any burnable absorber in the fuel rods.
7. The flooding is by pure water (no boron) at a temperature of 68°F. A limiting value of 1.0 gm/cm^3 is used for the density of water.
8. The 20.875 inch minimum center to center spacing is used for distance between all storage cells, (see Figure 1 on page 13). The actual distance between the outer assembly and the east wall is 54 inches. Only 30 inch distance between the outer assembly and the east wall is assumed in the analysis and this is conservative.
9. There are no absorber panels between the assemblies. Rack structure is only the L shaped inserts described in Table 2 on page 12.
10. All available storage cells are loaded with fuel assemblies.

A KENO model was set up using the above limiting fuel and rack parameters and resulted in a K_{eff} of 0.91295 with a 95 percent probability/95 percent confidence level uncertainty of +0.00130 ΔK.

Based on the analysis described above, the following equation is used to develop the maximum 95/95 K_{eff}:

$$K_{eff} = K_{worst} + B_{method} + \sqrt{ks_{worst}^2 + ks_{method}^2}$$

where:

K_{worst}	=	worst case KENO K_{eff}
B_{method}	=	method bias determined from benchmark critical comparisons
ks_{worst}	=	95/95 uncertainty in the worst case KENO K_{eff}
ks_{method}	=	95/95 uncertainty in the method bias

Substituting calculated values in the order listed above, the result is:

$$K_{eff} = (0.91295) + (0.0077) + \sqrt{0.00130^2 + 0.0030^2} = 0.92392$$

Since K_{eff} is less than 0.95 including uncertainties at a 95/95 probability/confidence level, the acceptance criterion for criticality is met for the Turkey Point Units 3 and 4 fresh fuel storage racks under full density water flooding conditions for storage of Westinghouse 15x15 fuel assemblies with maximum enrichments up to 4.55 w/o ^{235}U.

3.2 Low Density Optimum Moderation Analysis

The fresh fuel rack is analyzed for the low density optimum moderation condition under "worst case" scenario:

1. Maximum enrichment of 4.55 w/o ^{235}U
2. No pellet dishing
3. Fuel with 98% of theoretical density
4. Minimum assembly storage pitch of 20.875 inches.
5. Six feet thick concrete wall is assumed to surround the storage area on all sides which is conservative.
6. Top and bottom of the storage have 1 foot of water (at low density) followed by 6 feet of concrete.

The "worst case" scenarios conservatively account for fuel parameter variability and tolerances on rack dimensions (Table 2 on page 12). The KENO results for the "worst case" model are then used to develop the maximum 95/95 K_{eff} which is compared to the criticality safety limit of 0.98.

The following assumptions were used to develop the KENO model for the storage of fresh fuel in the Turkey Point Units 3 and 4 fresh fuel storage racks under low density optimum moderation condition:

1. The fuel assembly parameters relevant to the criticality analysis are based on the Westinghouse 15x15 DRFA design.
2. All fuel rods contain uranium dioxide at a maximum enrichment of 4.55 w/o in the entire length of the rod.
3. The fuel pellets are modeled assuming a UO_2 density which is 98% of theoretical density with no dishing fraction (0%) for the "worst case" conditions.
4. No credit is taken for any ^{234}U or ^{236}U in the fuel.
5. No credit is taken for any fuel grids.
6. No credit is taken for any burnable absorber in the fuel rods.
7. The moderator is low density water (no boron) at a temperature of 68°F. The optimum moderation typically occurs around 0.07 to 0.10 gm/cm^3 water density. Opti-

mum water density was determined and used in the analysis (See Figure 2 on page 14).

8. A minimum center to center spacing of 20.875 inches. See Figure 1 on page 13 and Table 2 on page 12 for details.
9. The entire rack is modeled and is assumed to be surrounded by concrete walls in all directions. For simplicity, the concrete wall thickness is assumed to be 72 inches. All available storage cells are loaded with fuel assemblies.

A KENO model was set up using the above limiting fuel and rack parameters. Water density of approximately 0.095 gm/cm^3 was found to lead to the highest reactivity (See Figure 2 on page 14). This density was used for the limiting calculation. This resulted in a K_{eff} of 0.83095 with a 95 percent probability/95 percent confidence level uncertainty of +0.00119 ΔK.

Based on the analysis described above, the following equation is used to develop the maximum 95/95 K_{eff}:

$$K_{eff} = K_{worst} + B_{method} + \sqrt{ks_{worst}^2 + ks_{method}^2}$$

where:

K_{worst}	=	worst case KENO K_{eff}
B_{method}	=	method bias determined from benchmark critical comparisons
ks_{worst}	=	95/95 uncertainty in the worst case KENO K_{eff}
ks_{method}	=	95/95 uncertainty in the method bias

Substituting calculated values in the order listed above, the result is:

$$K_{eff} = 0.83095 + (0.0077) + \sqrt{0.00119^2 + 0.0030^2} = 0.84188$$

This KENO model resulted in a K_{eff} of 0.84188 with a 95 percent probability/95 percent confidence level.

Since K_{eff} is less than 0.98 including uncertainties at a 95/95 probability/confidence level, the acceptance criteria for criticality is met for the Turkey Point Units 3 and 4 fresh fuel storage racks under optimum moderation condition.

Thus, for both full density water moderation and the optimum water density moderation the respective reactivity criteria are met. Westinghouse 15x15 fuel assemblies with maximum enrichments up to 4.55 w/o ^{235}U (with no axial blankets) meet the reactivity limits.

4.0 Discussion of Postulated Accidents in the Fresh Fuel Storage

Under normal conditions, the fresh fuel racks are maintained in a dry environment. The introduction of water into the fresh fuel rack area is the worst case accident scenario. The water flooding cases analyzed in this report are bounding accident situations which result in the most conservative fuel rack K_{eff}.

Other accidents can be postulated which could cause a reactivity increase in the fresh fuel racks and these are a fuel assembly drop on top of the rack and a fuel assembly drop between the rack and the wall. The fuel assembly drop between the rack and the wall is not possible for the Turkey Point Units 3 and 4 fresh fuel racks due to the construction configuration which precludes the drop of an assembly into any position other than a storage cell. For the fuel assembly drop on top of the rack, the double contingency principle[5] is applied. This states that assumption of two unlikely, independent, concurrent events is not required to ensure protection against a criticality accident. Thus, for the case of the fuel assembly drop on top of the rack; the absence of a moderator in the fresh fuel storage racks can be assumed as a realistic initial condition since assuming the presence of moderator would be a second unlikely, independent event.

Experience has shown that the maximum reactivity increase associated with a fuel assembly drop on top of the rack is less than 10 percent ΔK under the full density water condition.

Therefore, since the normal, dry fresh fuel rack reactivity for Turkey Point Units 3 and 4 is relatively low, less than 0.65, and the maximum reactivity increase for the fuel assembly drop on top of the rack for the dry condition would be much less than 10 percent ΔK, the maximum rack K_{eff} for the fuel assembly drop on top of the rack will meet the licensing bases.

5.0 Summary of Criticality Results

For the storage of Westinghouse 15x15 fuel assemblies in the Turkey Point Units 3 and 4 fresh fuel storage racks, the acceptance criteria requires the effective neutron multiplication factor to be $\leq$ 0.95, including uncertainties, for the fully flooded conditions and $\leq$ 0.98, including uncertainties, for the optimum moderation conditions.

This report shows that the acceptance criteria for criticality is met for the Turkey Point Units 3 and 4 fresh fuel racks for the storage of Westinghouse 15x15 fuel assemblies under both normal and accident conditions with enrichment limits tabulated below.

Fresh Fuel Storage	Storage of 15x15 fuel assemblies in all cells. Fuel assemblies must have a maximum enrichment no greater than 4.55 w/o in the entire length of the fuel rod.

The analytical methods employed herein conform to ANSI N18.2-1973, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants", Section 5.7 Fuel Handling System; ANSI 57.2-1983, "American Nuclear Society, American National Standard Design Requirements for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Plants", Section 6.4.2; ANSI/ANS-57.3-1983, "American Nuclear Society, American National Standard Design Requirements for New Fuel Storage Facilities at Light Water Reactor Plants"; ANSI 8.1-1983, "Validation of Calculational Methods for Nuclear Criticality Safety"; the NRC Standard Review Plan[4], Section 9.

Table 1. Nominal Fuel Parameters Employed in the Criticality Analysis

Parameter	Westinghouse 15x15 DRFA
Number of Fuel Rods per Assembly	204
Rod Clad O.D. (inch)	0.422
Clad Thickness (inch)	0.0243
Fuel Pellet O.D. (inch)	0.3659
Fuel Pellet Density (% of Theoretical)	96
Fuel Pellet Dishing Factor (%)	1.187
Rod Pitch (inch)	0.563
Number of Guide Tubes	20
Guide Tube O.D. (inch)	0.533
Guide Tube Thickness (inch)	0.017
Number of Instrument Tubes	1
Instrument Tube O.D. (inch)	0.533
Instrument Tube Thickness (inch)	0.017

Table 2. Fresh Fuel Storage Cell and Fuel Parameters

Parameter		Dimension	Tolerance	Value Used
Pitch	inch	21.0	+0.125/-0.125	20.875
Sourrounding Concrete	inch			72
L-angle Width	inch	2.5x2.5		2.5x2.5
L-angle Thickness	inch	0.25		0.25
L-angle ID	inch	9.0	+0.05/-0.075	9.0
Distance of wall from the Center of the Outer Assembly	inch			
West	inch			42
North	inch			30
East	inch			54
South	inch			73
Fuel Density	%	96	+2/-2	98
Dishing	%	1.187		0
Enrichment	%	4.50	+0.05/-0.05	4.55

Figure 1. Turkey Point Unit 4 Fresh Fuel Storage Cell Layout

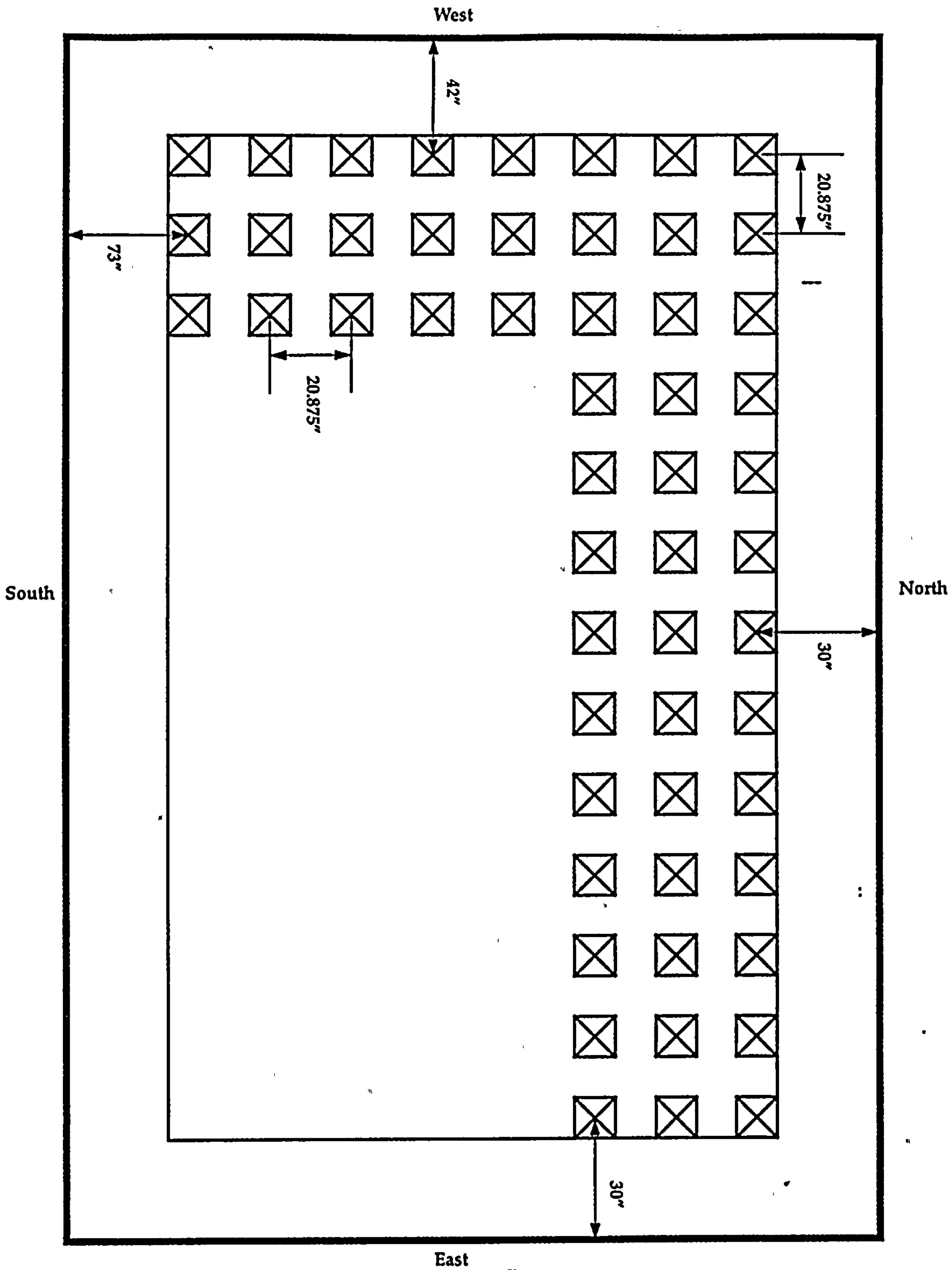
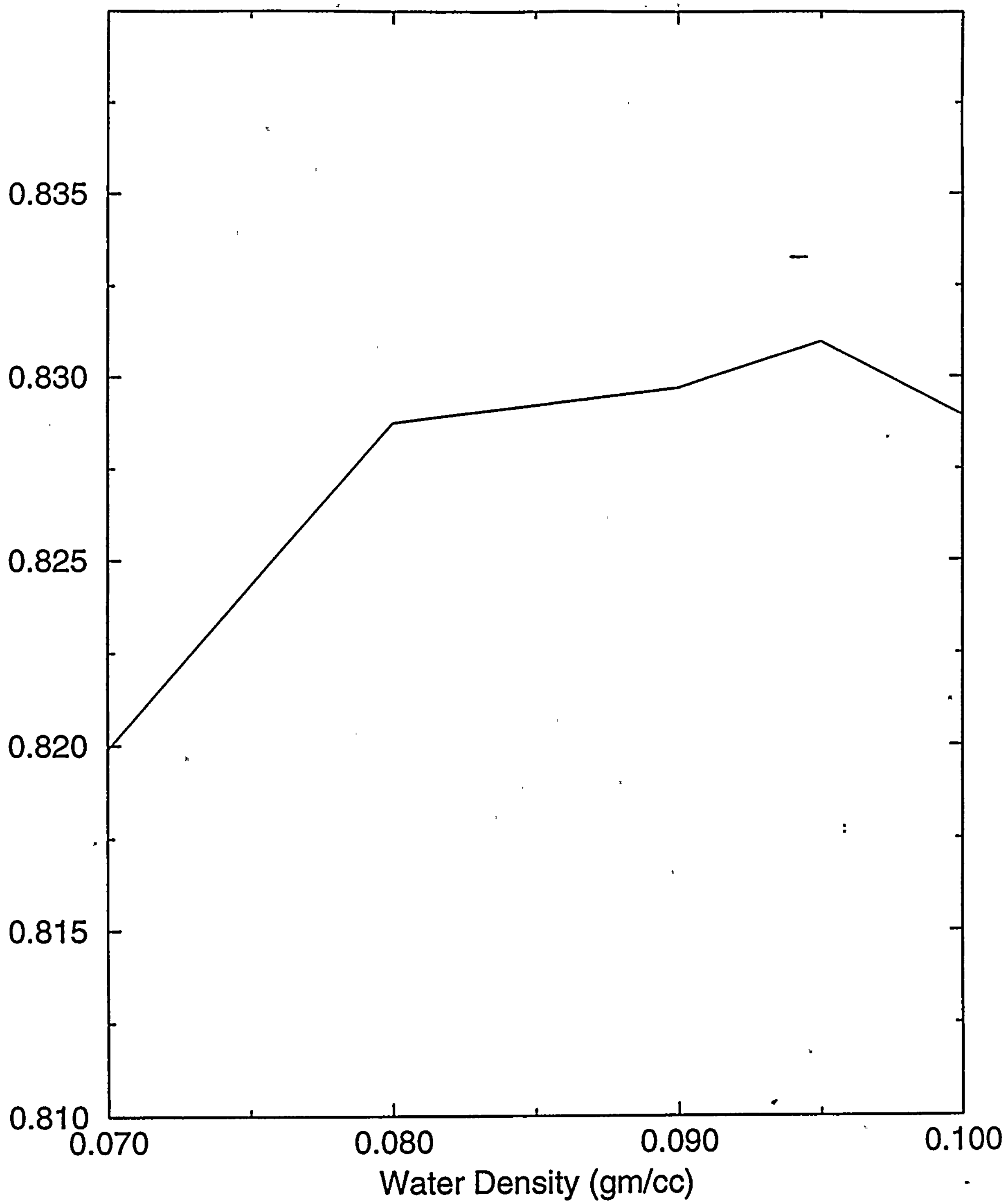


Figure 2. Turkey Point Units 3 and 4 Fresh Fuel Rack K_{eff} as a Function of Water Density



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ATTACHMENT 8

Turkey Point Units 3 & 4
Monthly Silica Concentration (ppm)

Table 1
Turkey Point Units 3 and 4
Monthly Silica Concentration (ppm)
January 1993 - September 1999

Month/YY	Unit 3	Unit 4
Jan-93	5.66	3.40
Feb-93	4.15	3.85
Mar-93	5.27	4.63
Apr-93	6.97	4.26
May-93	7.50	4.50
Jun-93	7.80	5.40
Jul-93	8.20	4.85
Aug-93	7.45	4.30
Sep-93	7.29	3.30
Oct-93	6.65	4.85
Nov-93	7.70	1.97
Dec-93	5.68	3.80
Jan-94	8.00	4.20
Feb-94	7.40	3.71
Mar-94	7.12	4.13
Apr-94	5.10	3.80
May-94	4.95	3.95
Jun-94	2.57	5.00
Jul-94	3.25	4.70
Aug-94	5.00	4.80
Sep-94	5.50	5.00
Oct-94	5.40	5.30
Nov-94	7.00	4.20
Dec-94	7.90	4.40
Jan-95	4.95	4.60
Feb-95	5.30	4.55
Mar-95	5.80	5.23
Apr-95	8.00	4.54
May-95	7.50	4.90
Jun-95	7.80	5.40
Jul-95	8.21	4.50
Aug-95	7.97	4.85
Sep-95	7.10	5.52
Oct-95	6.40	5.70
Nov-95	7.00	4.80
Dec-95	5.00	4.80

Month/YY	Unit 3	Unit 4
Jan-96	6.40	4.70
Feb-96	5.80	4.50
Mar-96	6.80	4.70
Apr-96	8.70	5.65
May-96	8.00	6.10
Jun-96	5.50	4.20
Jul-96	9.16	6.30
Aug-96	12.00	5.30
Sep-96	11.70	5.20
Oct-96	8.00	5.90
Nov-96	10.98	* 5.50
Dec-96	* 5.36	4.10
Jan-97	** 4.75	5.40
Feb-97	4.62	4.60
Mar-97	10.70	* 5.15
Apr-97	* 10.90	* 7.50
May-97	11.00	6.10
Jun-97	6.40	7.50
Jul-97	12.21	7.40
Aug-97	12.21	6.40
Sep-97	12.21	6.14
Oct-97	12.21	6.00
Nov-97	12.60	6.20
Dec-97	* 12.10	5.90
Jan-98	11.95	6.89
Feb-98	12.21	7.30
Mar-98	11.90	3.70
Apr-98	9.40	3.40
May-98	8.00	3.60
Jun-98	6.90	4.70
Jul-98	5.80	7.80
Aug-98	2.80	7.40
Sep-98	* 24.83	8.40
Oct-98	13.70	6.10
Nov-98	* 9.70	9.80
Dec-98	11.10	10.10

Month/YY	Unit 3	Unit 4
Jan-99	13.92	* 8.81
Feb-99	* 13.89	* 8.59
Mar-99	13.70	* 8.48
Apr-99	** 13.94	7.50
May-99	14.19	7.90
Jun-99	10.20	7.00
Jul-99	15.00	7.83
Aug-99	13.00	** 8.47
Sep-99	* 15.58	8.10

Notes:
* Average for month
** Interpolated

Data Sources:
Reference 17 and PTN Chemistry Logs