Florida Power & Light Company, 6501 South Ocean Drive, Jensen Beach, FL 34957

FPL

August 24, 2001

L-2001-199
10 CFR 50.90

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

RE: St. Lucie Unit 2
Docket No. 50-389
Proposed License Amendment Supplement
Containment Equipment Door and Containment
Airlock Doors Open During Core Alterations

On June 22, 2001 by letter L-2001-083 and pursuant to 10 CFR 50.90, Florida Power & Light Company (FPL) proposed to revise the St. Lucie Unit 2 Technical Specification (TS) 3.9.4, Containment Building Penetrations. TS 3.9.4.a. requires that the containment equipment door be closed during core alterations or movement of irradiated fuel within containment. TS 3.9.4.b. requires a minimum of one door in each airlock to be closed during core alterations or movement of irradiated fuel within containment. The proposed change to TS 3.9.4.a. would allow the containment equipment door to be open during core alterations and movement of irradiated fuel in containment provided: a) the equipment door is capable of being closed with four bolts within 30 minutes, b) the plant is in MODE 6 with at least 23 feet of water above the reactor pressure. The proposed change to TS 3.9.4.b. would allow both doors of each containment airlock to be open during core alterations and movement of irradiated fuel in containment provided: a) at least one door of each open containment airlock is capable of being closed, b) the plant is in MODE 6 with at least 23 feet of water above the reactor pressure vessel flange, and c) a designated individual is available outside each open containment airlock to close the door.

During the NRC review, the NRC project manager requested FPL to submit a clarification of previous fuel handling analyses (FHA). The requested clarification involved the terminology of original, current, and revised FHA presented in the safety analysis and the No Significant Hazard Consideration. Attachment 1 is a complete replacement and clarifies the description of the FHA of record and Safety Analysis in support of the proposed amendment. Attachment 2 is a complete replacement for the Determination of No Significant Hazard Consideration. Attachment 3 is a marked up showing the revised TS Bases change for NRC information. The wording of the proposed TS change is not changed by this submittal. The TS Bases will be revised under the plant's TS Bases Control Program. Attachment 4 of the original submittal (L-2001-083) was the revised fuel handling accident analysis, F-FSA-C-000001, Revision 0, *Determination of Fuel Handling Accident Radiological Releases in Support of Relaxation of St. Lucie Unit 2 Tech Spec 3.9.4*, prepared by Westinghouse Nuclear Systems and is unchanged.

A001

Approval of this proposed license amendment is requested by October 19, 2001 to support planning for the fall 2001 Unit 2 refueling outage (SL2-13).

Please contact us if there are any questions about this submittal.

Very truly yours,

Donald E. Jernigan
Vice President
St. Lucie Plant

DEJ/GRM

Attachments

cc: Regional Administrator, Region II, USNRC
Senior Resident Inspector, USNRC, St. Lucie Plant
Mr. William A. Passetti, Florida Department of Health

STATE OF FLORIDA)
) ss.
COUNTY OF ST. LUCIE)

Donald E. Jernigan being first duly sworn, deposes and says:

That he is Vice President, St. Lucie Plant, for the Nuclear Division of Florida Power & Light Company, the Licensee herein;

That he has executed the foregoing document; that the statements made in this document are true and correct to the best of his knowledge, information and belief, and that he is authorized to execute the document on behalf of said Licensee.

Donald E. Jernigan

STATE OF FLORIDA

COUNTY OF ST. LUCIE

Sworn to and subscribed before me

this 21 day of August, 2001
by Donald E. Jernigan , who is personally known to me.

Leslie J. Whitwell
Name of Notary Public - State of Florida

Leslie J. Whitwell
MY COMMISSION # DD020212 EXPIRES
May 12, 2005
BONDED THRU TROY FAIN INSURANCE, INC.

(Print, type or stamp Commissioned Name of Notary Public)

ATTACHMENT 1

SAFETY ANALYSIS

INTRODUCTION

Florida Power and Light Company (FPL) proposes to revise the St. Lucie Unit 2 Technical Specification (TS) 3.9.4, "Containment Building Penetrations." TS 3.9.4.a. requires that the containment equipment door be closed during core alterations or movement of irradiated fuel within containment. TS 3.9.4.b. requires a minimum of one door in each airlock to be closed during core alterations or movement of irradiated fuel within containment. The proposed change to TS 3.9.4.a. would allow the containment equipment door to be open during core alterations and movement of irradiated fuel in containment provided: a) the equipment door is capable of being closed with four bolts within 30 minutes, b) the plant is in MODE 6 with at least 23 feet of water above the reactor pressure vessel flange, and c) a designated crew is available at the equipment door to close the door. The proposed change to TS 3.9.4.b. would allow both doors of each containment airlock to be open during core alterations and movement of irradiated fuel in containment provided: a) at least one door of each airlock is capable of being closed, b) the plant is in MODE 6 with at least 23 feet of water above the reactor pressure vessel flange, and c) a designated individual is available outside each open airlock to close the door. Similar controls and procedures are already in place to support reactor coolant systems (RCS) operation at reduced inventory.

BACKGROUND

Technical Specification 3.9.4, "Containment Building Penetrations," requires the equipment door and at least one door in each containment airlock to be closed during core alterations and fuel movements (MODE 6). The basis for this requirement is to limit the effects of a fuel handling accident inside containment. The consequences of the fuel handling accident (FHA) for the reactor containment building is bounded by the effects of the accident occurring in the fuel handling building (FHB) since the FHB does not have an isolation system like the one installed in the reactor containment building. A reanalysis of the FHA was required with the assumption that the equipment door and all the containment airlock doors remain open for a two-hour period subsequent to the FHA.

FPL recalculated the doses resulting from the original design basis fuel handling accident incorporating the assumptions of Regulatory Guide (RG) 1.25, using bounding values for source term inventories, and assuming no credit for ventilation system filtration.

The original design basis fuel handling accident analysis occurring in the fuel handling building included the following major assumptions:

- For the limiting case, all the rods in one fuel assembly are damaged. The more realistic case has damage limited to 16 fuel rods from a single fuel assembly.

- In calculating the dose consequence, it is assumed that the incident occurs in the fuel handling building and no credit is taken for the FHB filtration system.

The results of this evaluation and the confirming analysis performed for NUREG-0843, Safety Evaluation Report Related to the Operation of St. Lucie Plant, Unit 2, Dated October 1981 were:

- Limiting Exclusion Area Boundary (EAB) Dose: 36 rem - thyroid and <1.0 rem - whole body
- Low Population Zone (LPZ) Dose: 15 rem - thyroid and < 1.0 rem - whole body

The original design basis fuel handling accident analysis was revised to support reracking of the Unit 2 spent fuel storage pool prior to the first refueling outage. This analysis was submitted by FPL letter L-84-47, dated March 13, 1984 and is the current analysis of record. This analysis included the following major assumptions:

- For the limiting case, all the rods in one fuel assembly are damaged. The more realistic case has damage limited to 16 fuel rods from a single fuel assembly.
- In calculating the dose consequence, it is assumed that the incident occurs in the fuel handling building and that the activity released triggers the airborne radiation monitors to isolate the normal fuel handling building ventilation system and automatically initiates the filtration systems.

The results of this evaluation were:

- Limiting Exclusion Area Boundary Dose: 3.0 rem - thyroid and 0.11 rem - whole body
- Low Population Zone Dose: 1.3 rem - thyroid and 0.046 rem - whole body

The NRC Safety Evaluation Report for the increased capacity of the Unit 2 spent fuel storage pool dated October 16, 1984 concluded that the increase in spent fuel storage capacity was acceptable since the resulting dose consequences were bounded by the original fuel handling accident doses.

REVISED DESIGN BASIS ANALYSIS

In support of the proposed license amendment, FPL is revising the design basis for the St. Lucie Unit 2 fuel handling accident analysis to include the effects of a fuel handling accident inside the reactor containment building. The dose calculations use the methodology of Regulatory Guide (RG) 1.25. In the revised analysis, the equipment door and/or all the containment airlock doors (the source is not bounded by the size of any opening) are assumed open with the refueling cavity filled with 23 feet of water above the reactor pressure vessel flange. The consequences of this event bound those from a fuel handling accident in the fuel handling building. The methodology use in calculating the control room doses is derived from an expression provided *in Nuclear Power Plant Control Room Ventilating System Design for Meeting General Design Criteria (GDC) 19*, 13th AEC Air Cleaning Conference, CONF740-807, Vol. 1, which determines the radiological doses based on an activity balance within the control room. Table 1 of Attachment 4 is the list of input parameters used in the fuel handling calculation.

The results of this re-analysis are as follows:

- Control Room Dose: 9.39 rem – thyroid and 0.02 rem – whole body
- Limiting Exclusion Area Boundary Dose: 61.6 rem – thyroid and 0.75 rem – whole body
- Low Population Zone Dose: 26.7 rem – thyroid and 0.33 rem – whole body

These values remain well within the acceptance criteria specified in NUREG-0800, "Standard Review Plan," Section 15.7.4, "Radiological Consequences of Fuel Handling Accidents." The EAB and LPZ inhalation thyroid doses are determined to be 61.6 rem and 26.7 rem, respectively. The EAB and LPZ whole body doses are calculated to be 0.75 rem and 0.33 rem, respectively. The NRC acceptance criteria on offsite doses are given in Reference 3 as 25% of 10 CFR 100 exposure guidelines, i.e., 75 rem for the thyroid dose and 6 rem for the whole body dose. Comparison of the results of the revised analysis against the acceptance criteria indicates that both of these criteria are met with more than adequate margin for both the EAB and the LPZ locations.

For the control room, the calculated inhalation thyroid dose is 9.39 rem and the whole body is 0.02 rem. The NRC acceptance criteria for control room habitability as provided in Section 6.4 in NUREG-0800 is 30 rem for inhalation thyroid dose and 5 rem for the whole body gamma dose. The results of the revised analysis for the control room doses indicate that these dose acceptance criteria are met with significant margins.

Assumptions used in this calculation are:

1. One whole fuel assembly is conservatively assumed damaged and its gap activity is assumed released to the water either in the reactor vessel or the spent fuel pool. This assumption is consistent with the recommendation of RG 1.25 (Reference 4).

2. The hottest fuel assembly with the highest radial peaking factor is assumed damaged. This assumption is consistent with the recommendation of RG 1.25 (Reference 4).

3. The overall decontamination factor for the iodine isotopes in the spent fuel pool and the reactor vessel is 100. This assumption is consistent with regulatory position C.1.g of RG 1.25 (Reference 4).

4. Minimum water depth between the damaged fuel assembly and the spent fuel pool or reactor cavity surface is 23 feet. This assumption is supported by St. Lucie Unit 2 Technical Specifications 3.9.10 and 3.9.11. These TS requirements satisfy the regulatory position in Section C.1.c of RG 1.25 (Reference 4).

5. All of the gap activity in the damaged fuel rods is assumed to be released and consists of:
   (a) 10% of all noble gases, except Kr-85
   (b) 30% of Kr-85
   (c) 10% of radioactive iodine, except I-131
   (d) 12% of I-131 in the rods at the time of the accident

   This assumption is consistent with regulatory position C.1.d of RG 1.25, (Reference 4) except for item (d). Item (d) uses a higher gap activity for I-131 isotope that is consistent with the guidance provided in NUREG/CR-5009 (Reference 8) for extended burn-up fuel use.

6. Fission product inventories are calculated assuming full power operation at the end of core life just before shutdown. A radial peaking factor of 1.65 is assumed. These assumptions are consistent with regulatory position C.1.e of RG 1.25 (Reference 4).

7. Iodine gas inventory is 99.75% inorganic and 0.25% organic. This assumption is consistent with regulatory position C.1.f of RG 1.25 (Reference 4).

8. The retention of noble gases in the pool is assumed to be negligible and therefore, a noble gas overall decontamination factor of 1 is used in the analysis. This assumption is consistent with regulatory position C.1.h of RG 1.25 (Reference 4).

9. For the exclusion area boundary (EAB) doses, the radioactive material that escapes from the spent fuel pool to the building is assumed to be released from the building over a two-hour time period. This assumption is consistent with regulatory position C.1.i of RG 1.25 (Reference 4).

10. Building exhaust system absorbers are not credited in the analysis. This is conservative in relation to regulatory position C.1.j of RG 1.25 (Reference 4).

11. No mixing of activity with fuel handling building air is assumed. This assumption is consistent with regulatory position C.1.k of RG 1.25 (Reference 4).

12. No credit is assumed for depletion of the effluent plume due to deposition or decay. This assumption conforms to regulatory position 3.a.(2) of RG 1.25 (Reference 4).

13. Consistent with the guidance of RG 1.25 (Reference 4), the following iodine isotopes are considered in the calculation of inhalation thyroid doses: I-131, I-132, I-133, I-134, and I-135. Of these, the contribution due to I-134 isotope are neglected due to the short half-life (52.6 min, from Reference 9) for this isotope.

14. The reactor would be subcritical for at least 72 hours prior to fuel movement before commencing refueling operations. This assumption is consistent with St. Lucie Unit 2 TS 3.9.3.

15. Control room intake and exhaust flow rates are assumed to be equal. The total in-leakage is assumed to be 450 cfm.

16. The location specific atmospheric dispersion factors that are provided in Reference 8 are assumed to be applicable for the EAB, low population zone (LPZ), and the control room.

17. A maximum average core burn-up of 41.35 GWD/MTU is assumed consistent with item 58 on page B-19 of Reference 7. This value corresponds to a maximum batch average discharge burn-up of 55 GWD/MTU consistent with item 102 on page B-26 of Reference 7. Since this batch is made up of assemblies that would be at burn-up levels higher and lower than this value, the peak assembly value is assumed to be at a higher value (about 58 GWD/MTU).

18. Only control room filters for filtering out iodine isotopes are considered in the analysis; no filtering in the containment or the fuel building is assumed in the analysis.

19. The dose conversion factors used in the analysis are consistent with those recommended in ICRP Publication II (Reference 10). These dose conversion factors are conservative relative to the TS 1.10 stipulated ICRP-30 thyroid dose conversion factors.

20. Part of the control room in-leakage (450-cfm) is assumed to be unfiltered (100 cfm) with the remainder (350-cfm) being filtered leakage. At the time of containment isolation on a containment isolation signal (CIS) (conservatively assumed to be 30 minutes after initiation of the event), the filtered in-leakage is assumed to be 0 cfm since the CIS would close the control room outside intake valves and start the control room booster fans. The booster fans recirculate the control room air through HEPA and charcoal filters at a rate of 2000 cfm in a closed loop. For control room gamma whole body and beta skin dose calculations, the unfiltered leakage is conservatively assumed to be the total in-leakage of 450 cfm. No filtering occurs for noble gases.

21. The fission product inventory calculation uses a multiplication factor of 30% on the activity calculated using the burn-up value in assumption 18 for additional conservatism.

22. The atmospheric dispersion factors used are those for ground level releases. These values are more conservative than those for elevated releases (see, for example, Figures 1 and 3 of RG 1.25 (Reference 4)). Note that releases from the containment equipment door are elevated releases and, as such, the atmospheric dispersion factors characteristic of these releases are expected to be smaller than the ground level release values.

The Updated Final Safety Analysis Report (UFSAR) will be revised and updated following the approval of this proposed license amendment to include the new design basis In-Containment Fuel Handling Accident Analysis.

DESCRIPTION OF THE PROPOSED CHANGE

FPL proposes to change the following Technical Specification in support of the proposed amendment.

1. TS 3.9.4 - Containment Building Penetrations: Revise the current TS 3.9.4 a. and TS 3.9.4 b. to read (with the proposed new requirements in bold).

   a. The equipment door closed and held in place by a minimum of four bolts, **or the equipment door may be open if:**

      1) it is capable of being closed with four bolts within 30 minutes,

      2) the plant is in MODE 6 with at least 23 feet of water above the reactor pressure vessel flange, and

      3) a designated crew is available at the equipment door to close the door.

   b. A minimum of one door in each airlock is closed, **or both doors of each containment airlock may be open if:**

      1) at least one door of each airlock is capable of being closed,

      2) the plant is in MODE 6 with at least 23 feet of water above the reactor pressure vessel flange, and

      3) a designated individual is available outside each open airlock to close the door.

2. Bases for Section 3.9.4: Revise the Bases for TS 3.9.4 to add the following paragraph.

   These restrictions include the administrative controls to allow the opening of both doors of each airlock (emergency and/or personnel) and the containment equipment door during CORE ALTERATIONS or movement of irradiated fuel in the containment provided that: a) at least one door of each airlock is capable of being closed; b) the plant is in MODE 6 with at least 23 feet of water above the reactor pressure vessel flange; c) a designated individual is available outside each open airlock to close the door; d) the equipment door can be closed with four bolts within 30 minutes; and e) an equipment door closure crew is available to close the equipment door. The capability to close the containment equipment door or the open containment airlocks include requirements that the equipment door or one of the airlock doors of each open airlock is capable of being closed and that any cables or hoses across the opening have quick disconnects to ensure the door is capable of being closed in a timely manner. The 30 minute closure time for the equipment

door is considered to start when the control room determines the need to establish containment integrity. This 30 minute assumption is significantly less than the 2 hour closure time assumed in the revised fuel handling accident analysis.

Justification

The proposed change to TS 3.9.4.a. would allow the containment equipment door to be open during core alterations and movement of irradiated fuel in containment provided: a) the equipment door is capable of being closed with four bolts within 30 minutes, b) the plant is in MODE 6 with at least 23 feet of water above the reactor pressure vessel flange, and c) a designated crew is available at the equipment door to close the door. The capability to close the containment equipment door includes the requirements that the door is capable of being closed and that any cables or hoses across the equipment door have quick-disconnects to ensure the door is capable of being closed in a timely manner. The proposed change to TS 3.9.4.b. would allow both doors of each containment airlock to be open during core alterations and movement of irradiated fuel in containment provided: a) at least one door of each open containment airlock is capable of being closed, b) the plant is in MODE 6 with at least 23 feet of water above the reactor pressure vessel flange, and c) a designated individual is available outside each open containment airlock to close the door. The capability to close a containment airlock door includes the requirement that the door is capable of being closed and that any cables or hoses across the airlock door have quick-disconnects to ensure the door is capable of being closed in a timely manner. Similar controls and procedures are already in place to support reactor coolant system (RCS) operation at reduced inventory.

The regulatory basis for TS 3.9.4, "Containment Building Penetrations," is to ensure that the primary containment is capable of containing fission product radioactivity that may be released following a fuel handling accident inside containment. This ensures that offsite radiation exposures are maintained well within the requirements of 10 CFR 100.

The purpose of the LIMITING CONDITION FOR OPERATION (LCO) is to minimize the release of radioactive material in the event of an in-containment fuel handling accident. Complying with the LCO assures that the assumptions reflected in the analysis for this accident as documented in the St. Lucie 2 UFSAR, Chapter 15.7.4.1.2, "Fuel Handling Accident" are met and the resulting doses are lower than calculated.

The original analysis of the fuel handling accident for St. Lucie Unit 2, assumed that the in-containment fuel handling accident was bounded by the fuel handling building accident. In that event, the entire amount of radioactivity released from the spent fuel pool is assumed to escape and that the activity released triggers the airborne radiation monitors to isolate the normal fuel handling building ventilation system and automatically initiates the filtration systems. The revised analysis estimates the dose with the containment equipment door and both doors of each containment airlock open. In the revised analysis, it is also assumed that the entire radioactivity released from the reactor cavity leaves the reactor containment building through the equipment door and both doors of each containment airlock, with no credit taken for filtration.

The proposed change contains restrictions on allowing the containment equipment door and both doors of each containment airlock to be open, provided that at least one door on each open containment airlock and equipment door will be available to perform its safety function. The restriction to be in MODE 6 with at least 23 feet of water above the fuel provides sufficient time to respond to a loss of shutdown cooling, ensures a minimum water level exists to provide sufficient shielding during fuel movement, and reduces the radioactivity released in the event of a fuel handling accident. The capability to close the containment equipment door and a door of each open containment airlock includes the requirement that the doors are capable of being closed and that any cables or hoses crossing through the doors have quick-disconnects to ensure the doors are capable of being closed in a timely manner. Requiring that a designated individual be available to close the equipment door and a door of each open containment airlock following evacuation of the containment will minimize the release of radioactive material. Administrative requirements will be established for the responsibilities and appropriate actions of the designated individuals in the event of an in-containment fuel handling accident. These requirements will include the responsibility to be able to communicate with the control room, responsibility to ensure that the doors are capable of being closed in the event of an in-containment fuel handling accident, door closure, and to implement single containment airlock door open operations in the event of a fuel handling accident. These administrative controls will ensure refueling containment integrity would be established in the event of an in-containment fuel handling accident.

The revised calculations and analysis indicate that the basis for the Technical Specification requirements will be met with the equipment door and both doors of each containment airlock open during core alterations with the ability to close the equipment door and one door on each open containment airlock following a FHA.

EVALUATION

Containment Integrity

Technical Specification 3.6.1, "Containment Integrity," requires that containment integrity be maintained while in MODES 1 to 4. During MODES 1 to 4, the reactor coolant system contains significant energy that provides the motive force for the expulsion of radionuclides subsequent to a design basis accident (DBA). This TS allows the opening of containment vessel penetrations under administrative control. The relaxation described in this evaluation is being sought for MODE 6 where the effects of a fuel handling accident inside containment are the event of concern and are bounded by the DBA.

Containment Closure

Technical Specification 3.9.4, "Containment Building Penetrations," requires that a minimum of one door on each open containment airlock, the equipment door, as well as other containment penetrations (except as permitted under Administrative Controls), be closed during core alterations or movement of irradiated fuel within the containment. This requirement is more conservative than the assumptions used in the revised St. Lucie Unit 2 Updated Final Safety Analysis Report (UFSAR), Chapter 15.7.4.1.2, Fuel Handling Accident. The revised accident analysis assumes that, in the event of a fuel handling accident in containment, all of the iodine and noble gases that become airborne within the containment are assumed to escape and reach the site boundary and low population zone with no credit taken for the containment building barrier or for decay or deposition. The revised fuel handling accident analysis also assumes a minimum water level of 23 feet above the top of the fuel in the core and a minimum post-reactor shutdown decay time of 72 hours prior to fuel movement.

During a refueling outage, other work inside containment does not stop during fuel movement or core alterations. Licensed operators moving the reactor fuel are in constant communications with the control room and are procedurally required to inform the control room that the containment evacuation alarm be sounded in the event of a fuel handling accident. The personnel inside the reactor containment building will evacuate. This requires that personnel operate the personnel airlock doors to exit the containment. The revised analysis assumes that the reactor cavity water does not delay the dispersion of the source term gases following the accident. This is a conservative assumption when considering the dose to plant personnel inside containment. The plant personnel inside the reactor containment building would have adequate time to evacuate prior to the source term gases dispersing inside the reactor containment building which has a free volume of 2.5 million cubic feet. In MODE 6, "Refueling" the reactor coolant system is depressurized and there is no system active to pressurize the reactor containment building during a FHA. Therefore, the effects of a radioactive release in MODES 1 through 4 from a pressurized RCS would have a greater effect since the reactor containment building would become pressurized.

The containment emergency airlock opens into the fuel handling building, which has an air filtration system that releases through a monitored plant vent stack. The opening of these doors will allow control element assembly extension shafts to be passed directly from the containment to the fuel handling building or from the fuel handling building into containment. The extension shafts are normally stored in containment until refueling containment integrity is no longer required by TS. This creates an unnecessary radioactive source inside containment for this period of time. Elimination of the extension shaft storage will reduce personnel exposure of the plant workers near the storage area.

The containment equipment door will have a closure crew available to close this door. The closure crew is trained for timely equipment door closure. The door can be closed without electrical power available and within 20 minutes of notification. The equipment door closure crew currently provides this function during RCS reduced inventory operations in accordance with FPL commitments made as part of Generic Letter (GL) 88-17.

From a practical standpoint, the current TS 3.9.4 will not prevent all radioactive releases from the containment following a fuel handling accident. There may be a number of people in containment during a refueling outage, even during fuel movement and core alterations. Should a fuel handling accident occur inside containment, the airlock doors would be cycled several times to evacuate personnel from containment. With each containment airlock cycle, more containment air would be released. Under the proposed change, the containment could be evacuated more quickly with timely refueling integrity being established subsequently. This would reduce dose to workers.

Control Room Ventilation

The UFSAR discusses St. Lucie Unit 2 compliance with GDC 19. The NRC Safety Evaluation of the St. Lucie Plant Unit No. 2, dated October 1981, concluded that the proposed design of the control room and the ventilation system would meet GDC 19 criteria. The St. Lucie Unit 2 control room is designed with an emergency cleanup system, which is actuated by a containment isolation actuation signal (CIAS) from either unit or a control room outside air intake (CROAI) high radiation signal. The filter trains filter a portion of the recirculated air. Outside air make up and toilet and kitchen exhaust flows are isolated by butterfly valves actuated by a CIAS (either unit) or CROAI high radiation signal. Later, a reduced outside air flow, filtered by the cleanup part of the system, is manually adjusted to maintain a positive pressure in the control room which prevents the ingress of unfiltered (i.e., potentially contaminated) outside air.

The CIAS was designed to control the radioactive release from the plant under accident conditions such as a loss of coolant accident (LOCA). Since the doses conservatively calculated in the event of a LOCA event are significantly higher than the doses calculated in the event of an in-containment fuel handling accident, the requirements of GDC-19 are satisfied. The control room dose is bounded by the large break LOCA. The results of the revised analysis for the fuel handling accident indicate that the LOCA dose is still the bounding accident for the control room dose.

Applicable Regulatory Requirements/Criteria

NUREG-0800, "Standard Review Plan", Section 15.7.4, "Radiological Consequences of Fuel Handling Accidents," describes the acceptance criteria for this event as, "the calculated doses at the exclusion boundary are well within the exposure guidelines of 10 CFR Part 100. 'Well within' shall mean 25% or less of 10 CFR Part 100, i.e., 75 rem to the thyroid and 6 Rem for the whole-body doses." Neither the current nor the revised design basis fuel handling accident analysis takes credit for the containment building barriers. The results of the calculations performed (Attachment 4, page 16) show that the offsite dose consequences of a fuel assembly dropped inside containment are well within the 10 CFR Part 100 limits. Therefore, the proposed change does not result in a significant hazard.

U. S. NRC Regulatory Guide 1.25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors," is NRC guidance which describes a method acceptable to the NRC staff for licensee evaluation of the potential radiological consequences of a fuel handling accident. Thc parameters of concern and the acceptance criteria applied are based on the requirements of 10 CFR 100 with respect to the calculated radiological consequences of a FHA and GDC 61 with respect to appropriate containment, confinement, and filtering systems.

NUREG/CR-5009, "Assessment of the Use of Extended Burn-up Fuel in Light Water Power Reactors," relates to the expected release fraction for the radioactive iodine. According to this report, the calculated release fraction for extended burn-up fuel may be up to 20% higher than that assumed in Regulatory Guide 1.25 for iodine 131.

The methodology, assumptions, and results of the revised FHA with the proposed Technical Specification changes comply with the applicable regulatory requirements, criteria, and guidance.

10 CFR Part 50, Appendix A, General Design Criteria

GDC 16, "Containment Design," requires that reactor containment and associated systems shall be provided to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment and to assure that the containment design conditions important to safety are not exceeded for as long as the postulated accident conditions require.

GDC 19, "Control Room," requires that a control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including loss-of-coolant accidents. Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident. Equipment at appropriate locations outside the control room shall be provided (1) with a design capability for prompt hot shutdown of the reactor, including necessary instrumentation and controls to maintain the unit in a safe condition during hot shutdown, and (2) with a potential capability for subsequent cold shutdown of the reactor through the use of suitable procedures.

GDC 54, "Piping Systems Penetrating Containment," requires that piping systems penetrating primary reactor containment shall be provided with leak detection, isolation, and containment capabilities having redundancy, reliability, and performance capabilities which reflect the importance to safety of isolating these piping systems. Such piping systems shall be designed with a capability to test periodically the operability of the isolation valves and associated apparatus, and to determine if valve leakage is within acceptable limits.

GDC 56, "Primary Containment Isolation," describes the isolation provisions that must be provided for lines that connect directly to the containment atmosphere and which penetrate primary reactor containment unless it can be demonstrated that the isolation provisions for a specific class of lines are acceptable on some other defined basis.

GDC 61, "Fuel Storage and Handling and Radioactivity Control," requires that the fuel storage and handling, radioactive waste, and other systems which may contain radioactivity shall be designed to assure adequate safety under normal and postulated accident conditions.

The assumptions and results of the revised FHA analysis, coupled with the proposed Technical Specification changes, demonstrate compliance with the above GDCs.

EVALUATION CONCLUSIONS

Based on review of the licensing bases documentation and the results of the reanalysis of the fuel handling accident inside the reactor containment building, it is concluded that the proposed license amendment is acceptable and that code requirements are maintained.

REFERENCES

1. St. Lucie Unit 2, Updated Final Safety Analysis Report, Amendment 13.

2. St. Lucie Unit 2, Technical Specifications, Amendment 115.

3. NUREG-0800, US NRC Standard Review Plan

4. Regulatory Guide 1.25, "Assumptions Used for Evaluating the Potential Radiological Consequences of Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors," 1972.

5. Murphy, K. G. and Campe, K. M., "Nuclear Power Plant Control Room Ventilating System Design for Meeting General Design Criteria 19," 13th AEC Air Cleaning Conference, CONF740-807, Vol. 1, pp. 401-430.

6. NUREG/CR-3011, "Dose Projection Considerations for Emergency Conditions at Nuclear Power Plants," Stoetzel, G.A., et al, May 1983.

7. SL2-FE-0181, Rev. 09, "St. Lucie Unit 2 EPAC," H. F. Jones, Jr., October 23, 1998.

8. NUREG/CR-5009, "Assessment of the Use of Extended Burnup Fuel in Light Water Power Reactors," D. A. Baker, et al, February 1988.

9. "Table of Isotopes," Edited by Lederer, C. M., and Shirley, V. S., 7th Edition, John Wiley & Sons, New York, 1978.

10. ICRP Publication II, "Recommendations of the International Commission on Radiological Protection," A Report of Committee 11 on Permissible Dose for Internal Radiation (1959), Pergamon Press, New York, 1960.

11. Regulatory Guide 1.109, Rev. 1, "Calculation of Annual Dose to Man From Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50," Appendix I, October 1977.

12. Westinghouse Nuclear Systems Analysis Number L-FSA-C-000001, "Determination of Fuel Handling Accident Radiological Releases in Support of Relaxation of St Lucie Unit 2 Tech Spec 3.9.4," Dated 3/01/01.

ATTACHMENT 2

DETERMINATION OF NO SIGNIFICANT HAZARD CONSIDERATION

Description of Proposed License Amendments

Florida Power and Light Company (FPL) proposes to revise Technical Specification 3.9.4, "Containment Building Penetrations." TS 3.9.4.a. requires that the containment equipment door be closed during core alterations or movement of irradiated fuel within containment. TS 3.9.4.b. requires a minimum of one door on each airlock to be closed during core alterations or movement of irradiated fuel within containment. The proposed change to TS 3.9.4.a. would allow the containment equipment door to be open during core alterations and movement of irradiated fuel in containment provided: a) the equipment door is capable of being closed with four bolts within 30 minutes, b) the plant is in MODE 6 with at least 23 feet of water above the reactor pressure vessel flange, and c) a designated crew is available at the equipment door to close the door. The capability to close the containment equipment door includes the requirements that the door is capable of being closed and that any cables or hoses across the equipment door have quick-disconnects to ensure the door is capable of being closed in a timely manner. The proposed change to TS 3.9.4.b. would allow both doors of each containment airlock to be open during core alterations and movement of irradiated fuel in containment provided: a) at least one door on each open containment airlock door is capable of being closed, b) the plant is in MODE 6 with at least 23 feet of water above the reactor pressure vessel flange, and c) a designated individual is available outside each open containment airlock to close a door. The capability to close the containment airlock door includes the requirement that the door is capable of being closed and that any cables or hoses across the airlock door have quick-disconnects to ensure the door is capable of being closed in a timely manner.

Introduction

The Nuclear Regulatory Commission provides standards for determining whether a significant hazards consideration will exist (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 amendment.

Discussion

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

The proposed change to TS 3.9.4 would allow the containment equipment door and both doors of each containment airlock to be open during fuel movement or core alterations. Currently, the equipment door is closed with four bolts and a single door on each containment airlock is closed during fuel movement or core alterations to prevent the escape of radioactive material in the event of an in-containment fuel handling accident. Neither the containment equipment door nor either of the containment airlock doors is an initiator of an accident. Whether the containment equipment door or both doors of the containment airlocks are open or closed during fuel movement and core alterations has no affect on the probability of any accident previously evaluated.

Allowing the containment equipment door and the containment airlock doors to be open during fuel movement or core alterations does not significantly increase the consequences from a fuel handling accident. The calculated offsite doses are well within the limits of 10 CFR Part 100. In addition, the calculated doses are larger than the expected doses because the calculation does not incorporate the closing of the containment equipment door or the containment airlock doors after the containment is evacuated, which is much less than the two hours assumed in the analysis. The proposed change should significantly reduce the dose to workers in containment in the event of a fuel handling accident by reducing the time required to evacuate the containment.

The changes being proposed do not affect assumptions contained in other plant safety analyses or the physical design of the plant, nor do they affect other Technical Specifications that preserve safety analysis assumptions. Therefore, operation of the facility in accordance with the proposed amendment would not involve a significant increase in the probability or consequences of an accident previously analyzed.

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

The proposed change to Technical Specification 3.9.4, "Containment Building Penetrations," affects a previously evaluated fuel handling accident. The new Fuel Handling Accident Analysis assumes that all of the iodine and noble gases that become airborne escape and reach the exclusion area boundary and low population zone with no credit taken for filtration, the containment building barrier, or for decay or deposition. Since the proposed change does not involve the addition or modification of equipment nor does it alter the design of plant systems and the revised analysis is consistent with the Fuel Handling Accident Analysis, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The margin of safety as defined by 10 CFR Part 100 has not been significantly reduced. The calculated dose is well within the limits given in 10 CFR Part 100 or NUREG 0800. The proposed change does not alter the bases for assurance that safety-related activities are performed correctly or the basis for any Technical Specification that is related to the establishment of or maintenance of a safety margin. Therefore, operation of the facility in accordance with the proposed amendment would not involve a significant reduction in a margin of safety.

Summary

Based on the above discussion, FPL has determined that the proposed amendment 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 change does not involve a significant hazards consideration as defined in 10 CFR 50.92.

Environmental Impact Consideration Determination

The proposed license amendment changes requirements with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The proposed amendment involves no significant increase in the amounts and no significant change in the types of any effluents that may be released offsite, and no significant increase in individual or cumulative occupational radiation exposure. FPL has concluded that the proposed amendment involves no significant hazards consideration and therefore, meets the criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), an environmental impact statement or environmental assessment need not be prepared in connection with issuance of the amendment.

ATTACHMENT 3

St. Lucie Unit 2 Marked-Up Technical Specification Bases Page

B 3/4 9-1

For NRC Information Only

3/4.9 REFUELING OPERATIONS

BASES

3/4.9.1 BORON CONCENTRATION

The limitations on reactivity conditions during REFUELING ensure that: (1) the reactor will remain subcritical during CORE ALTERATIONS, and (2) a uniform boron concentration is maintained for reactivity control in the water volumes having direct access to the reactor vessel. These limitations are consistent with the initial conditions assumed for the boron dilution incident in the safety analyses. The value specified in the COLR for K_{eff} includes a 1% delta k/k conservative allowance for uncertainties. Similarly, the boron concentration value specified in the COLR includes a conservative uncertainty allowance of 50 ppm boron.

3/4.9.2 INSTRUMENTATION

The OPERABILITY of the startup neutron flux monitors ensures that redundant monitoring capability is available to detect changes in the reactivity condition of the core.

3/4.9.3 DECAY TIME

The minimum requirement for reactor subcriticality prior to movement of irradiated fuel assemblies in the reactor pressure vessel ensures that sufficient time has elapsed to allow the radioactive decay of the short lived fission products. This decay time is consistent with the assumptions used in the safety analyses.

3/4.9.4 CONTAINMENT BUILDING PENETRATIONS

The requirements on containment penetration closure and OPERABILITY ensure that a release of radioactive material within containment will be restricted from leakage to the environment. The OPERABILITY and closure restrictions are sufficient to restrict radioactive material release from a fuel element rupture based upon the lack of containment pressurization potential while in the REFUELING MODE.

3/4.9.5 COMMUNICATIONS

The requirement for communications capability ensures that refueling station personnel can be promptly informed of significant changes in the facility status or core reactivity condition during CORE ALTERATIONS.

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These restrictions include the administrative controls to allow the opening of both doors of each airlock (emergency and/or personnel) and the containment equipment door during CORE ALTERATIONS or movement of irradiated fuel in the containment provided that a) at least one door of each airlock is capable of being closed; b) the plant is in Mode 6 with at least 23 feet of water above the reactor pressure vessel flange; c) a designated individual is available outside each open airlock to close the door; d) the equipment door can be closed with four bolts within 30 minutes; and e) an equipment door closure crew is available to close the equipment door. The capability to close the containment equipment door or the open containment airlocks include requirements that the equipment door or one of the airlock doors of each open airlock is capable of being closed and that any cables or hoses across the opening have quick disconnects to ensure the door is capable of being closed in a timely manner. The 30 minute closure time for the equipment door is considered to start when the control room determines the need to establish containment integrity. This 30-minute assumption is significantly less than the 2-hour closure time assumed in the revised fuel handling accident analysis