Florida
Power
CORPORATION
Crystal River Unit 3
Docket No. 50-302

August 28, 1996
3F0896-18

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D.C. 20555-0001

Subject: Technical Specification Change Request No. 208, Revision 0

Reference: NRC to FPC letter 3N0589-25, dated May 25, 1989

Dear Sir:

Florida Power Corporation (FPC) hereby submits Technical Specification Change Request No. (TSCRN) 208 requesting amendment to Operating License No. DPR-72. As part of this request, the TSCRN and the proposed new Technical Specification and Bases pages are provided.

The TSCRN proposes to establish a new Technical Specification Limiting Condition for Operation applicable to the Control Complex Habitability Envelope (CCHE). The CCHE encloses the control room and other Control Complex spaces served by the Control Complex Emergency Ventilation System. The components of the CCHE include walls, floors, floor drains, roof, doors, isolation dampers and penetration seals. The proposed Technical Specification will allow breaches to exist in the CCHE for limited periods of time, provided personnel and materials are available to close the breach if notified by the control room. The NRC approved FPC's control room habitability strategy (NUREG-0737, Item III.D.3.4) in the referenced letter.

FPC's request is based, in part, on the low probability of the occurrence of events which would potentially challenge the CCHE, and on the low safety significance of having breaches in the CCHE for relatively short, controlled periods of time. The specific challenges are either 1) large releases of radioactive material such as would occur following a loss of coolant accident with core damage or a fuel handling accident involving irradiated fuel, or 2) a release of toxic gas from an onsite storage location. A deterministic evaluation is provided which illustrates the minimal effect on plant and public safety represented by approval of the proposed Technical Specification. A probabilistic safety analysis is also provided which quantifies the risk associated with allowing breaches in the CCHE, and demonstrates that there will be no significant impact on risk.

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This TSCRN is being submitted because none of the existing Technical Specifications explicitly address breaches in the CCHE. Approval of this TSCRN will establish clear guidance for control of CCHE breaches. Accordingly, we are providing the following proposed schedule to support timely review, dialogue, and approval of this TSCRN:

Initial Status Meeting (or teleconference)	September 30, 1996
Acceptance Review	October 18, 1996
Status Meeting	November 13, 1996
Status Meeting	December 20, 1996
Amendment Approval	February 15, 1997

We would appreciate your review and concurrence with this schedule or ask that you advise us of any necessary changes. If questions or concerns arise during your review regarding the technical content of the submittal or the proposed schedule, please contact Brian Gutherman, Manager, Nuclear Licensing, at 352-563-4566.

Sincerely,

P. M. Beard, Jr.
Senior Vice President
Nuclear Operations

PMB/SCP

Attachment

xc: Regional Administrator, Region II
Senior Resident Inspector
NRR Project Manager

STATE OF FLORIDA

COUNTY OF CITRUS

P. M. Beard, Jr. states that he is the Senior Vice President, Nuclear Operations for Florida Power Corporation; that he is authorized on the part of said company to sign and file with the Nuclear Regulatory Commission the information attached hereto; and that all such statements made and matters set forth therein are true and correct to the best of his knowledge, information, and belief.

P. M. Beard, Jr.
Senior Vice President
Nuclear Operations

P. M. Beard, Jr., personally known to me. Subscribed and sworn to before me, a Notary Public in and for the State and County above named, this 28th day of August, 1996.

LYNNE S. SMITH
Notary Public (print)

Notary Public (signature)

LYNNE S. SMITH
MY COMMISSION # CC 514380
EXPIRES: December 18, 1999
Bonded Thru Notary Public Underwriters

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

IN THE MATTER)
) DOCKET NO. 50-302
FLORIDA POWER CORPORATION)

CERTIFICATE OF SERVICE

P. M. Beard, Jr. deposes and says that the following has been served on the Designated State Representative and Chief Executive of Citrus County, Florida, by deposit in the United States mail, addressed as follows:

Chairman,
Board of County Commissioners
of Citrus County
Citrus County Courthouse
Inverness, FL 34450

Administrator,
Radiological Health Services
Department of Health and
Rehabilitative Services
1323 Winewood Blvd.
Tallahassee, FL 32301

A copy of Technical Specification Change Request No. 208, Revision 0.

FLORIDA POWER CORPORATION

P.M. Beard, Jr.
Senior Vice President
Nuclear Operations

P. M. Beard, Jr., personally known to me. Sworn to and subscribed before me this 28th day of August 1996

LYNNE S. SMITH
Notary Public (print)

Notary Public (signature)

LYNNE S. SMITH
MY COMMISSION # CC 514380
EXPIRES: December 18, 1999
Bonded Thru Notary Public Underwriters

FLORIDA POWER CORPORATION
CRYSTAL RIVER UNIT 3
DOCKET NO. 50-302/LICENSE NO. DPR-72
REQUEST NO. 208, REVISION 0
CONTROL COMPLEX HABITABILITY ENVELOPE

LICENSE DOCUMENT INVOLVED: Technical Specifications

PORTIONS: Specification 3.7.18

DESCRIPTION OF REQUEST:

This change request proposes to establish a new Limiting Condition for Operation (LCO) which provides a period of time during which breaches in the Control Complex Habitability Envelope (CCHE) may exist that create leakage paths larger than authorized in the control room habitability calculations. FPC Engineering Calculation I-92-0011 (Revision 1) determined a total of 43 square inches of 'unidentified' breach may exist in the envelope continuously without the control room operator radiation dose exceeding regulatory limits should a design basis accident occur. It is proposed that breaches in excess of the design basis may exist for up to 24 hours per occurrence, provided that personnel and materials are immediately available to close the breach if notified by control room operators.

REASON FOR REQUEST:

The principal function of the CCHE is to provide an enclosed environment from which the plant can be operated following an uncontrolled release of radioactivity or toxic gas. The design of the CCHE for radiation protection is based on the radioactivity release associated with a design basis loss of coolant accident (LOCA). The source terms used in the design basis LOCA analysis are consistent with Regulatory Guide 1.4, Revision 2, June 1974. The limiting event for the CCHE due to a toxic gas release is a catastrophic rupture of the Sulfur Dioxide (SO_2) storage tank at the adjacent coal fired power generation units. Both of these are extremely low probability events.

The current Control Room Emergency Ventilation System (CREVS) Technical Specification (TS), 3.7.12, addresses the components of the ventilation system loop including fans, dampers, filters, and associated ductwork. That TS is silent on the other components that make up the protected envelope including the walls, doors, roof, floors, and floor drains of the Control Complex (CC), and does not explicitly address the isolation dampers. This specification is being proposed to provide controls on the existence of CCHE breaches, and provide a reasonable response to their discovery or creation. Based on the low probability of the occurrence of either of the postulated threats to the CCHE, existence of a CCHE breach in excess of design limits for a limited period of time would not result in an undue safety risk to the public.

EVALUATION OF REQUEST:

Description of the Analysis Basis

In MODES 1-4 the 30-day thyroid dose is the limiting case as determined by FPC Engineering Calculation I-86-0003, and results from airborne radioiodine. In MODES 5 and 6 the threat of toxic gas limits the allowed amount of outside air leakage into the Control Complex. CCHE breaches in excess of design limits in MODES 1-4 affect radiation dose to the control room operators more significantly than they affect toxic gas concentrations in the control room. This is due to a variation in the system's response to the two principal hazards.

At the onset of the design basis radiological event, a loss of coolant will cause pressure to rise in the Reactor Containment Building (RB). Upon sensing pressure in excess of 4 pounds per square inch (gauge) in the RB, the Engineered Safeguards (ES) system will initiate isolation of the CREVS. This will close the CREVS isolation dampers and isolate the CCHE from the outside atmosphere. The CCHE will be isolated well before a radiological release begins from the RB. Therefore the entry of airborne radioiodine will be limited to leakage through breaches in the CCHE.

The threat from toxic gas is an SO_2 tank rupture at the adjacent coal fired units with immediate creation of a highly concentrated plume of toxic gas. Given the worst environmental conditions, the plume drifts toward the CC and is drawn into the open CREVS intake. No credit is taken for notification of a release from the adjacent fossil plant or alarm of the local toxic gas monitors at the tank since those monitors are non-safety related and non-ES powered, although these administrative controls and alarms exist. Once SO_2 is drawn into the CREVS intake, monitors therein sense the gas and initiate closure of the isolation dampers. By the time dampers close, sufficient SO_2 is postulated to have been drawn in to create uncomfortable conditions for the control room operators. However these conditions will not be life threatening. The symptoms created will be sufficient to prompt operators to don self contained breathing apparatus (SCBA) devices to escape the discomfort.

Deterministic Risk Evaluation

The design basis calculation for post accident control room dose uses the radiological source term applicable for the design basis LOCA. This source term is consistent with the guidance of Regulatory Guide 1.4, Revision 2, June 1974. This calculation shows that the integrated thirty day gamma dose and the equivalent thyroid dose to the control room operators will remain within the limits of 10 CFR 50 Appendix A, General Design Criterion 19 (GDC 19). This scenario bounds all other analyzed events for control room doses in MODES 1-4. Inoperability of the CCHE or CREVS is not a contributor to core damage frequency for any evaluated accident. Since the CCHE is only effective in limiting dose following a significant accident, the existence of breaches does not increase the probability of occurrence of any evaluated event.

The CREVS, when operating in the emergency (recirculation) mode, maintains essentially zero differential pressure between the CCHE and the outside environment. The CREVS fully isolates the CCHE from the outside and no make up or discharge air is designed to pass across the envelope boundary. The only source of outside air during this mode of operation is infiltration through

breaches that may exist in the boundary. The assumed differential pressure to drive air infiltration is 1/8 inch water gauge.

The existence of a breach of the CCHE will be limited to 24 hours duration by the proposed TS. This will effectively limit the scope of planned work activities which create breaches, and the time required for resealing the CCHE. Assuming the existence of a breach and the capability ready to seal the breach as required by the proposed TS ACTIONS, operator dose is not expected to reach or exceed GDC 19 limits if a radiological release were to occur. GDC 19 dose limits are based on a thirty day integrated exposure. Therefore, sealing a breach within 1 hour provides reasonable assurance that dose limits will not be exceeded.

Breaches in the CCHE do not significantly increase the threat that control room operators will be incapacitated by SO_2. The threat from an SO_2 tank rupture is a sudden increase in CC atmosphere concentration of SO_2 due to an undetected highly concentrated plume of gas being drawn into the CCHE prior to CREVS isolation. Once isolation is accomplished, concentrations will not increase further by a significant amount due to the existence of breaches.

The analysis of various SO_2 accident scenarios is contained in FPC Engineering Calculation I-87-0005, Revision 2. In the limiting scenario, SO_2 concentrations will rise to uncomfortable levels (15 ppm) in the control room in approximately 10 seconds and reach maximum concentrations (29 ppm) in less than 20 seconds. The Protective Action Limit in Standard Review Plan 6.4, "Habitability Systems" for a one to two minute exposure to SO_2 is 36 ppm. By the time the maximum concentration is reached in the control room, dampers will have already closed as a result of SO_2 detector actuation and the CCHE will be isolated. The initial highly concentrated portion of the plume would quickly move past the CREVS boundary and a lower concentration plume would continue to be present for many hours after the tank rupture. The initial plume concentration is calculated to be approximately 10,000 ppm of SO_2. The plume concentration after one minute is 19 ppm and after five minutes is 7 ppm. Therefore, minutes after CREVS isolation, air leaking into the CCHE would have a lower concentration of SO_2 than the enclosed atmosphere.

In the scenario of tank rupture with CREVS already isolated, SO_2 concentration in the CC would reach 5 ppm (detectable by the senses but not uncomfortable) in 4 minutes and would peak at 13.5 ppm at 17 hours. These lower concentrations would not increase the threat to the operators or to their ability to respond to plant conditions. Therefore, the existence of breaches in the CCHE while isolated would not change the conclusions of the existing evaluation.

Probabilistic Risk Evaluation

A probabilistic risk evaluation of the challenges to the CCHE was performed using the current CR-3 Probabilistic Safety Assessment (PSA) model, following the guidelines given in the "PSA Application Guide," published by the Electric Power Research Institute (EPRI). To evaluate the risk of a radiological release of the magnitude used in the dose analysis, the probability of a core damage accident with early containment failure was modeled. This event was chosen since it is the only accident scenario during power operation which results in a significant control room dose and it bounds the dose for a fuel handling accident occurring during shutdown MODES. The estimated frequency of such a sequence is $3.1X10^{-7}$ per year. If it were assumed that the full allowed outage time of 24 hours was exercised twice per month every month during the year (a very conservative assumption) the risk of a radiological threat occurring during this period is $5.09X10^{-8}$. This is a very low occurrence rate and does not represent a significant increase in the threat to the health of the operators.

To determine the radiological risk to the public from a toxic gas release, a scenario was modeled which estimated the frequency of core damage due to an SO_2 tank rupture. There is no direct cause and effect relationship between an SO_2 tank rupture and core damage. To create core damage, an SO_2 tank rupture must cause incapacitation of control room operators, a transient must occur which is not adequately mitigated by automatic safety systems and to which the operators cannot respond, and the transient must precipitate core damage prior to the arrival of an operations relief shift.

A tank rupture is assumed to occur which vaporizes approximately 16% of the tank inventory immediately, leaving the remaining volume in an open pool to boil off to the surrounding atmosphere (the boiling point of SO_2 liquid is 14°F). Failure of an individual tank penetration will not result in sufficiently high airborne concentrations to create a threat to CR-3 control room operators, and can be effectively mitigated by emergency response teams from the adjacent coal fired units.

The following paragraphs describe the complex series of events that were assumed in evaluating core damage risk due to a hypothetical SO_2 tank rupture. First, of course, an SO_2 tank rupture must take place. Three modes of SO_2 tank rupture were examined:

1) tank rupture due to normal actuation of the heaters, followed by failure of the heaters to shut off,

2) tank rupture due to spurious actuation of the heaters, followed by failure of the heaters to shut off, and

3) random tank failure.

Tank rupture due to heater actuation can only occur if the thermostat, pressure switch, and relief valve also fail to function. Since these components are a mix of electrical and mechanical safety devices, no common mode failure is plausible. For mechanisms 1) and 2) above, the probabilities of tank failure were computed to be $1.7X10^{-5}$ and $2.9X10^{-9}$ per year respectively. The frequency of random tank rupture computed from industry data is $2.1X10^{-4}$ per year. Summing the contributions from these three mechanisms results in a tank rupture frequency of $2.3X10^{-4}$ per year.

Assuming a tank rupture, the wind has to be favorable to transport the SO_2 gas toward the CR-3 CREVS outside air intake. Historic wind direction information in the CR-3 FSAR, Figure 2-10, shows the probability of wind direction favorable for transport is 0.4. Next, the operators would have to fail to don respiratory equipment before being incapacitated by the SO_2 fumes. Operator failure to don respirators is quite improbable since the gas can be readily detected by the senses of taste and smell at very low concentrations (1 to 3 parts per million) and due to the extremely irritating nature of the gas.

The probability of operator incapacitation would be a function of the amount of time available for operators to don respirators before significant concentrations of SO_2 reach the control room. There are two SO_2 alarms available to warn the operators. One is associated with the detectors located adjacent to the SO_2 tank and the other is associated with the detectors in the CR-3 intake duct. In the event neither of the alarms work, there is still time between the operator's detection of the SO_2 and his or her incapacitation. This is because the calculated maximum SO_2 concentration will be below the Protective Action Limit, which is established so that an individual exposed for up to two minutes will quickly recover after donning a fresh air respirator. There is also the consideration that all of the operating crew would have to be overcome to assume that the control board is unmanned. The probability that all of the operating crew would be overcome given that both of the SO_2 alarms failed was conservatively assumed to be 1.0. The probability that all of the operating crew would be overcome given that only the SO_2 alarm at the CR-3 intake duct worked was assumed to be 0.1. The probability that all of the operating crew would be overcome given that both alarms worked was assumed to be 0.05. The probability of each alarm failing was assumed to be 0.01. The assumed probability of any or all of the operating crew being overcome by SO_2 is very conservative considering the highly irritating nature of SO_2 and the low detection threshold by the human senses.

If all of the operating crew are overcome, the reactor does not immediately proceed toward core damage. A transient must occur and the automatic safety systems must fail to mitigate the transient before a relief crew arrives. The CR-3 PRA initiating event data base was used to estimate the conditional probability of a transient occurring given that the SO_2 tank has ruptured, and was computed to be $5.75X10^{-3}$. The probability of the automatic mitigating systems failing was conservatively assumed to be 0.1.

A fault tree model was constructed for the sequence of events whereby an SO_2 tank rupture could precipitate a core damage event, and this frequency was calculated to be $2.7X10^{-9}$ per year. This increase in core damage frequency falls well within the non-risk-significant region of the "quantitative screening criteria" for "permanent changes impacting core damage frequency" given in the "PSA Application Guide." If this frequency were assumed to be low by a factor of ten, it would still not meet NRC's criterion for accident sequence reporting for the Individual Plant Examinations (IPE) requested via Generic Letter 88-20. This criterion directs that all core damage sequences with a frequency equal to or greater than $1X10^{-7}$/year be reported, and that additional actions may be required only when frequencies are two or more orders of magnitude higher. As a relative measure of the SO_2 tank rupture risk significance, the CR-3 core damage frequency for all internal events is $8.3x10^{-6}$/year, over three orders of magnitude higher than the SO_2 tank rupture core damage frequency.

The probabilities for human errors and failure of mitigating systems to operate were conservatively assumed as indicated above. A second calculation was performed to assess the sensitivity of the SO_2 tank rupture core damage frequency of $2.7X10^{-9}$ per year to the assumed probabilities. In the sensitivity analysis each of the probabilities for operators failing to don respirators and being overcome by toxic gas regardless of alarms were assumed to be 1.0. Additionally the probability of automatic safety systems failing to mitigate a transient for a period of time sufficient to allow a relief crew to arrive was also set to 1.0. The combined effect of these changes resulted in a frequency of core damage due to SO_2 tank rupture of $5.3X10^{-7}$ per year. This increase in core damage frequency is still well within the non-risk significant region of the quantitative screening criteria for permanent changes to the plant in the PSA Application Guide.

This probabilistic risk evaluation demonstrates that approval of this specification would not have a significant impact on either the control room operator dose or on the risk to the public.

ENVIRONMENTAL IMPACT EVALUATION

Radiological Evaluation

While 10 CFR 51 requires an environmental assessment (EA) or environmental impact statement (EIS) for any "major Federal action significantly affecting the quality of the human environment," it does allow the NRC discretion in evaluating the extent to which EA's or EIS's are necessary. EA's or EIS's are not required for any action included in the list of "categorical exclusions" set forth in 10 CFR 51.22(c). Specifically, 10 CFR 51.22(c)(9), provides that an EA is not required for the issuance of an amendment provided that:

(i) the amendment involves no significant hazards consideration,

(ii) there is no significant change in the types or significant increase in the amounts of any effluents that may be released offsite, and

(iii) there is no significant increase in individual or cumulative occupational radiation exposure.

FPC considers that the provisions of 10 CFR 51.22(c)(9) are applicable to this request for a Technical Specification LCO for the CCHE. For the reasons described in this submittal, FPC believes that the three criteria of 10 CFR 51.22(c)(9) are satisfied. Therefore, this Technical Specification amendment should be considered under the "categorical exclusions" provisions of 10 CFR 51.22(c)(9). There will be no environmental impact from allowing breaches to exist in the CCHE. For the reasons given in this submittal that there will be no change in offsite consequences due to this action, its impact is bounded by the impacts assumed in the existing Final Environmental Statement (FES) for CR-3. Even if the NRC chooses to perform an EA, information provided in the FES, together with this submittal should assist the NRC in making a "finding of no significant impact" in accordance with 10 CFR 51.32.

Non-Radiological Evaluation

The non-radiological environmental concerns for the Crystal River Energy Complex that impact CR-3 are discharge canal water temperature and circulating water flow. FPC has considered the non-radiological effects of this Technical Specification amendment, and has determined that there will be no change in the CR-3 circulating water flow rate or discharge temperature. Therefore, no changes are required in the Environmental Protection Plan because of this action.

SHOLLY EVALUATION OF REQUEST

Florida Power Corporation has reviewed the requirements of 10 CFR 50.92 as they apply to the proposed allowance for breaches in the Control Complex Habitability Envelope (CCHE) in excess of the design limit for no more than 24 hours per occurrence, with contingencies in place to seal breaches, and considers the allowance not to involve a significant hazards consideration. In support of this conclusion, the following analysis is provided:

1. Operation of the facility in accordance with the proposed amendment would not involve a significant increase in the probability of occurrence or consequences of an accident previously evaluated. The CCHE is designed to limit the radiation dose to the control room operating staff following an accident involving massive reactor core damage and breach of the reactor containment building. Since the CCHE is only effective in limiting dose following an accident, the existence of breaches would not significantly increase the probability of occurrence of any evaluated event. The features of the CCHE have no direct function in mitigating the offsite consequences of any evaluated accident. It is desirable to limit dose to the control room operators, however exceeding 30 day integrated doses during a design basis event would not hinder normal operator response, or increase the offsite consequences.

   A second design objective of the CCHE is the protection of the control room operating staff from excessive exposure to toxic gas. Although the toxicity of chlorine or sulfur dioxide presents a personnel hazard, escape or detection of toxic gas is not part of any design basis event for reactor or containment transients. A release of toxic gas would not initiate any evaluated event. As described elsewhere in this submittal, the existence of additional breaches in the CCHE would not significantly increase the probability of operator incapacitation. Even in the event of operator incapacitation, the consequences of the design basis LOCA would not be increased.

   Based on the foregoing, the proposed amendment does not significantly increase the probability or consequence of an accident previously evaluated.

2. The proposed time period for the existence of CCHE breaches does not create the possibility of a new or different kind of accident from previously evaluated accidents. During the period of time when CCHE breaches are greater than the design calculation, there exists the possibility of increased control room operator dose. However, the existence of a breach and the capability ready to seal the breach within 1 hour as required by the proposed TS ACTIONS, provides reasonable assurance that 30 day integrated operator dose will not reach or exceed GDC 19 limits if a radiological release were to occur. Since the expected dose will remain within GDC 19 limits, the proposed amendment will not create the possibility of a new or different kind of accident. Likewise in the case of a toxic gas release, no new type of accident different from those previously evaluated could occur as a result of the existence of CCHE breaches. As described previously in this submittal, their existence would not lead to significantly higher toxic gas exposure of control room operators.

Based on the foregoing, the proposed amendment does not create the possibly of a new or different kind of accident.

3. The proposed amendment does not involve a significant reduction in a margin of safety. The existence of additional breaches in the CCHE for short periods of time does not significantly increase the risk of control room operator exposure to radiation or toxic gas. There is no change in the risk to the public since the CCHE has no direct function in mitigating the offsite consequences of any evaluated accident. Any event that could create these exposures has an extremely low probability of occurrence, and while the potential for higher exposure exists if additional breaches are open, the short duration allowed would not significantly increase the risk of exposure. Probabilistic evaluations have been conducted which verify this conclusion. Therefore, the existing margin of safety would not be reduced.