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

December 2, 1994
3F1294-02

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

Subject: Report of 10 CFR 50.59 Safety Evaluations

Dear Sir:

Florida Power Corporation (FPC) is submitting the attached report in accordance with the provisions of 10 CFR 50.59(b)(2). This report provides a listing of the changes to Crystal River Unit 3 (CR-3) as described in the Final Safety Analysis Report (FSAR) which required safety evaluations. The safety evaluations are included as required.

Sincerely,

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

PMB/JWT:ff

Attachment

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

070038

IE47

In the following discussions, Yes or No answers are provided to the following questions:

1. Is the probability of occurrence of an accident previously evaluated in the FSAR increased?

2. Are the consequences of an accident previously evaluated in the FSAR increased?

3. Is the possibility of an accident of a different type than any previously evaluated in the FSAR created?

4. Is the probability of occurrence of malfunction of equipment previously evaluated in the FSAR increased?

5. Are the consequences of malfunction of equipment previously evaluated in the FSAR increased?

6. Is the possibility for malfunction of equipment of a different type than any previously evaluated in the FSAR created?

7. Is the margin of safety, as defined in the basis for any Technical Specification, reduced?

In some of the attached material, there may be only 3 answers to the 10 CFR 50.59 questions. These 10 CFR 50.59 evaluations were performed before FPC began using the NSAC-125 guidelines. In these cases, the numbers refer to the following questions:

1. Is the probability of an occurrence or the consequences of an accident or malfunction of equipment important to safety, as previously evaluated in the FSAR increased?

2. Is the possibility of an accident or malfunction of a different type than any previously evaluated in the FSAR created?

3. Is the margin of safety, as defined in any Technical Specifications, reduced?

MAR 93-06-16-01
ATWS Setpoint Change

1. No, the additional indication provides no mechanism for accident initiation, and the change in enable setpoint reduces the potential for plant upsets and thus accident occurrence.

2. No, the additional indication provides no increase in accident consequences because no credit has been taken for them in accident mitigation. The increased enable setpoint for AMSAC has been shown acceptable up to 50% power by analysis and the new setpoint is below this limit (instrument error included).

3. No, the instrumentation used are existing and the added indication provides nothing new for accident mitigation. The increase in AMSAC enable setpoint provides no mechanism for a different type of event.

4. No, the added instrumentation indication and AMSAC setpoint change are isolated such that failure cannot propagate back to the Class 1E part of the circuit. Therefore, this cannot cause critical equipment malfunction.

5. No, the added indication and AMSAC setpoint change are non-safety related functions and as such do not affect consequences of events or failures.

6. No, the added instrumentation indication and AMSAC setpoint change are non-safety related functions isolated from critical equipment so no additional possibility of equipment failure is added.

7. No, the added instrumentation indication and AMSAC setpoint change are not addressed by Tech Specs. Therefore, no margin of safety is impacted. ATWS margin to analysis limits have been shown to be acceptable.

MAR 93-08-03-01
Secondary Services Heat Exchanger Replacement

1. No, the equipment being replaced is non-safety related and is being replaced with equipment that is slightly larger in capacity, but performing the same design function.

2. No, the new equipment performs the same design function. The design function of the equipment is to remove heat from secondary system components which are not used to reduce the consequences of an accident.

3. No, the equipment does not affect safety related components. Components and systems which are affected by this equipment can not affect safety related components.

4. No, the new equipment performs the same design function as the original components. Therefore, no increase in the probability exists. The new equipment performs more effectively than the original equipment.

5. No, the equipment will not affect safety aspects of the plant. No scenario analyzed in the FSAR could be made worse by the malfunction of this equipment.

6. No, the design requirements and function of the original components have been retained in the new components. No new aspects of operation have been introduced.

7. No, the Secondary Services heat exchangers are not covered by Tech Specs.

MAR 92-10-02-01
Transfer Tube Gate Valve & Upender Limit Switches Relocation

1. No, the limits switches used to position interlocks for upender baskets and transfer tube valve were located underwater. To increase reliability and ease of maintenance, the limit switches were moved above the water line.

2. No, the function of each limit switch remains the same. Use of a mechanical linkage will actuate each limit switch. Failure of the mechanical linkage will result in no fuel transfer.

3. No, with the relocation of the limit switches comes additional hardware in the form of linkage. The design of the mechanical linkage allows for a fail-safe operation, i.e., if linkage were to break, carriage travel will stop.

4. No, the mechanical linkage is the new part of the design that will allow the limit switches to be moved above water. The linkage will be under constant tension to keep it in place. Safeguard provisions remain the same. This design will interrupt the power to the transfer carriage if the mechanical linkage were to break.

5. No, failure of the limit switches stops fuel transfer. This will not increase the consequences of malfunctions identified in the FSAR Fuel Handling Accident. This modification maintains the same component functions.

6. No, FSAR Chapter 14 has analyzed a complete failure of a fuel assembly during refueling operations. With the fail-safe mode of mechanical linkage break, fuel transfer stops. This modification does not create a different accident scenario.

7. No, fuel transfer limit switches are not used to mitigate an accident or bring the plant to a safe shutdown condition.

Problem Report 94-0231
Atmospheric Dump Valves Flow Capacity

1. No, Atmospheric Dump Valves (ADV) are not credited with causing any accident currently evaluated in the FSAR. A change in capacity would have no impact on the probability of any accident occurring.

2. No, for the OTSG Tube Rupture accident, release of radioactivity was evaluated for an ADV capacity of 418,500 lbm/hr at 600°F and 900 psig. By decreasing this capacity, the consequences are reduced, not increased.

3. No, by decreasing the ADV capacity, the Main Steam Safety Valves (MSSV) will be challenged earlier and possibly more frequently. This situation does not have the potential to create a new type of accident since failure of a MSSV to actuate properly (either fail to open or fail to close) have been already evaluated in the FSAR.

4. No, decreasing the ADV capacity has no impact on the functioning of the valve, only on the response pressure/temperature curves during transients. This in turn would have no impact on equipment function. Failure of the ADVs is not addressed.

5. No, the consequences of a malfunction of the ADVs is not addressed in the FSAR. Therefore, decreasing the ADV capacity has no impact on the FSAR evaluations.

6. No, no equipment function being impacted by decreasing the ADV capacity. The turbine bypass valves (TBV) and the MSSVs may be actuated earlier in the transient, however, this will not cause a malfunction of the ADVs.

7. No, adequate relief capacity exists even without the ADvs. This position is supported by analysis. Analysis also supports reduced capacity for either ADVs or TBVs. Since reduced capacity would reduce total release through the ADvs during a STGR, margin would not be reduced.

Problem Report 93-0218
Operating Time for Reactor Building Spray

A FPC engineering calculation documented the acceptability of revising the response time (to full flow) for the Reactor Building Spray System from 71 to 90 seconds. This increase in RB Spray response time resulted in an increase in peak containment accident pressure from 53.9 to 54.2 psig. Post-LOCA dose rates were previously evaluated and determined to be acceptable for RB spray response times of 120 seconds and less.

1. No, MAR 88-05-24-01 moved the building spray pumps from block 1 to block 6 of the emergency diesel generator loading sequence. As a result of this modification, the BS pumps will start later than previously evaluated. The MAR did not consider the effect of RB pump start time on RB pressure. The increased time prior to BS initiation and attainment of full flow results in a new peak containment pressure following the design basis LOCA. The resulting peak pressure of 54.2 psig remains less than the containment design pressure of 55 psig. As such, the probability of occurrence of an accident previously evaluated in the FSAR is not increased.

2. No, the consequences of an accident previously evaluated in the FSAR are not increased as a result of this change. The proposed resultant RB pressure of 54.2 psig remains less than the containment design pressure of 55 psig, ensuring no adverse affect on the fission product barrier (containment). The issue of increased dose as a result of the longer response time has been previously evaluated under a separate change to the FSAR. Since the previous dose evaluation was made assuming a response time of 120 seconds, the 90 seconds proposed as part of this change is conservative.

3. No, the possibility of an accident of a different type than any previously evaluated in the FSAR is not created by the increased BS system response time and new RB pressure of 54.2 psig. This conclusion is based on the fact there is no change to the design function or purpose of the BS System or containment. As such, no new credible failure mode is created.

4. No, the probability of occurrence of malfunction of equipment previously evaluated in the FSAR is not increased. The safety analysis still assumes the failure of one ECCS train and an increase in RB Spray response time has no affect on this probability. Maintaining post-accident calculated peak containment pressure less than the design value of 55 psig ensures the probability of containment failure is not increased above that previously evaluated.

5. No, the consequences of malfunction of equipment previously evaluated in the FSAR are not increased since there is no change in the design function of the equipment previously evaluated. Only the time frame for performing the function is affected.

Problem Report 93-0218
Operating Time for Reactor Building Spray
(Continued)

6. No, the possibility of malfunction of equipment of a different type than previously evaluated in the FSAR is not created because no changes are being made to the basic function of previously evaluated equipment.

7. No, the margin of safety as defined in the basis for any Technical Specification is not reduced since peak containment pressure of 54.2 remains less than the design value of 55 psig. The value of 55 psig represents the accepted margin of safety for containment operability. BS Response times which yield RB pressures less than 55 psig and offsite doses within Part 100 acceptance criteria do not decrease the margin of safety defined in the basis for the Improved Technical Specifications (ITS). A parallel ITS Bases change is being processed to accurately reflect new RB Spray response times and peak post-accident containment pressure.

Deletion of Commitment to Submit Startup Report

FSAR Section 1. 12 requires a special report (the Startup Report) be submitted to the NRC following any one of a number of identified changes to the plant design. The content of the report includes a description of the testing performed to validate acceptability of the change as well as the results of this testing. This inf ormation is submitted to the NRC after resumption of power operation, such that prior NRC review and approval prior to power operation is not required. In reality, the report is nothing more than a convenient compilation of the subject testing program, provided to the NRC for an after-the-fact review, if they are so inclined. Given that the test descriptions are already located in FSAR Chapter 13 and applicable plant procedures, and that these two documents along with the test results are quality records (available and retrievable for review), the requirement to submit a separate report to describe these activities is considered to be a significant administrative burden without commensurate benefit. Any problems with predicted versus measured test results would be processed in accordance with the CR-3 corrective action system (which is also a quality record). As such, the requirement to compile and submit the startup report is deleted.

1-6. No, the startup report is an administrative requirement which has no effect on the operation of the plant. The testing which is the subject of the report will continue to occur regardless of whether the report is compiled and submitted to the NRC.

7. The startup report is not addressed within Technical Specifications. As such, deleting the requirement to compile and submit the report does not reduce the margin of safety as defined in the basis for any Technical Specification.

POQAM Procedures CH-383, CH-322, & EM-307
Post Accident Sampling System

1. No, these procedure changes in no way affect any design basis accident evaluation or increases the likelihood of any design basis occurring as addressed in the FSAR.

2. No, these procedure changes do not affect any barrier performance that could result in an increase of offsite or onsite doses occurring during an accident evaluated in the FSAR.

3. No, these procedure changes in no way affects or causes system upsets, transients, or radiation releases which could create the possibility of any type accident.

4. No, these procedure changes do not affect or change any safety-related equipment function or performance as addressed in the FSAR.

5. No, these procedure changes do not affect or change any safety-related equipment function or performance as addressed in the FSAR.

6. No, these procedure changes do not address or affect any equipment function or performance that could place the plant in an unsafe condition.

7. No, these procedure changes will not affect the safety margin of any Technical Specification.

POQAM Procedures SP-135A, SP-135B, & SP-135C
Engineered Safeguards Actuation System Channel Response Time Tests

1. No, REA 92-1601 addresses the issue that the 2.1 second EDG voltage recovery time is conservative and was based on an out-of-date calculation. This recovery time is required for MCC starters to achieve the required 85% voltage to pickup during Block 1. Analyses have determined that the voltage recovery to 100% voltage is less than 1 second for Block 1. Therefore, a recovery time of 1 second to achieve the 85% voltage is conservative.

2. No, these revisions do not change the testing method or equipment lineups as it presently exists. Instead, it ensures that the system response are within limits assumed in the safety analyses.

3. No, these revisions provide assurance that the ESAS action required for each channel is completed within the time assumed in the safety analyses.

REA 93-1320
Bypass Makeup System Prefilters

1. No, the bypassing of the prefilters or the replacement of the prefilters with prefilters with a higher micron rating will not prevent the proper function and operation of the Make Up and Purification System. The MU System currently may be temporarily operated with the prefilters and post filters in the bypass mode in the event these filters were to clog. Any increase in demineralizer resin or post filter replacement which may be required as a result of bypassing the prefilters or replacing the existing prefilters with a larger micron rated prefilter will be detected by current procedural sampling techniques performed to determine the time for replacement of these components. Differential pressure alarms are also currently installed which alarm when the differential pressure within the prefilters or post filters is not within established limits prior to the necessity for filter replacement. Hardware modifications, with the exception of potentially replacing the current 2 micron prefilter with a larger micron rated filter such as the 10 micron prefilter previously used, are not required which will maintain the system as currently designed assuring the safety and seismic design of the system. As a result, maintaining the current safety and seismic design and procedural sampling controls for the MU System as currently established, the probably of an accident previously evaluated in the FSAR is not increased by operating the MU System with the prefilters in the bypassed mode or by replacing the current 2 micron prefilters with larger rated micron filters such as the 10 micron prefilters previously used.

2. No, the MU System safety and seismic design as well as the procedural sampling techniques will be maintained when operating the MU System with the prefilters bypassed or with the prefilters replaced with prefilters with a larger micron rating, thus assuring the proper function of the MU System. Consequences of an accident as previously evaluated within the FSAR will not be affected by operating the MU System in this mode as system design and function will be maintained.

3. No, the current safety and seismic design of the MU system will be maintained. The current configuration for the MU System provides for temporary operation of the system with the prefilters bypassed. Administrative controls have been established and will be maintained to assure sampling of the prefilters (if a larger micron rated filter replaces the existing 2 micron filter), post filters and demineralizer resins is conducted to determine when these components require replacement, assuring the proper function of the MU System with the prefilters bypassed or with the installation of larger micron rated replacement prefilters. Since the function and safety and seismic design of the system will be maintained, the possibility of an accident different than any previously evaluated in the FSAR will not be created.

REA 93-1320
Bypass Makeup System Prefilters
(Continued)

4. No, administrative procedure controls have been established to assure adequate sampling occurs to determine the need for replacement of the demineralizer resins, prefilters (if a larger micron prefilter is used to replace the current 2 micron prefilter) or the post filters, assuring the function of the MU System. An increase in post filter, prefilter (if the prefilters are replaced with a larger micron rated filter) or demineralizer resin replacements will be detected by these administrative controls assuring the proper function of the MU System. As a result, the probability or malfunction of equipment as previously addressed in the FSAR will be maintained.

5. No, the current configuration of the MU System provides for temporary operation of the MU System with the prefilters bypassed. Permanently bypassing the prefilters or replacing the current 2 micron prefilters with a larger micron rated filter will not adversely effect the function of the MU System. Administrative controls have been established and will be maintained to assure adequate sampling occurs to determine when these components require replacement to assure proper function of the MU System. As a result, the consequences of a malfunction of equipment as previously evaluated in the FSAR will be maintained.

6. No, administrative controls for sampling of the MU System to determine when the prefilters (if the prefilters are replaced with larger micron rated filters), post filters or demineralizer resins need replacement and the safety and seismic design of the MU System will be maintained by operating the system with the prefilters bypassed or by replacing the prefilters with a larger micron rated filter. Clogging of these components would be detected by the administrative controls currently established. Administrative controls currently established for sampling chloride and fluoride levels will be maintained assuring these limits will not be exceeded (to include Technical Specification limits), preventing potential stress corrosion cracking of stainless steel components within the RC and MU Systems. Additional filters, MUFL-3A and MUFL-3B, are currently installed within the MU System downstream of the post filters, assuring seal water provided to the RC Pump seals will be within acceptable limits preventing damage or premature replacement of the RC Pump Seals. Demineralizer resin bed channeling would be detected via the current administrative sampling controls which require monitoring of the differential pressure and radiation levels across and within the prefilters, post filters and demineralizers assuring the proper function of these components. As a result, the possibility for a malfunction of equipment of a different type than previously evaluated within the FSAR is not created by bypass operation of the prefilters or replacement of the prefilters contained within the MU System with a larger micron rated filter.

REA 93-1320
Bypass Makeup System Prefilters
(Continued)

7. No, the design or operation of the MU System prefilters is not addressed in any Technical Specification.

MAR 90-08-014-01A
Reactor Building Purge Exhaust System

1. No, the modification to the control logic of the Reactor Building Purge Exhaust System consisted of bypassing the non-environmentally qualified components (temperature switch, smoke detector and deluge valve) which control the above mentioned auto stop functions. Additionally, the fans start permissives requiring full open position of their associated dampers (limit switches) and the local control station for operating the fans are bypassed.

   The following components were added to the control system: (1) the addition of 3-way hand valves which allows the bypass of the solenoid valves under emergency conditions, thereby allowing the ability to control the position of the dampers (AHD-94, AHD-95, AHD-96, AHD-97 and AHD-98) for the Air Handling Fans (AHF-7A and AHF-7B) and (2) the addition of key switches to bypass (defeat) the permissive fan start/run controls, such that the fans will still be able to operate during Post-LOCA conditions by bypassing electrical equipment which is not environmentally qualified.

   None of the Fire Protection System is environmentally qualified. Due to the installation of this modification, the Post-LOCA operation of the Reactor Building Purge Exhaust System (Hydrogen Purge) will not be shut down on detection of Products of Combustion (smoke) or high temperature in the discharge duct, since the components which are used to detect smoke and high temperature in the duct are not environmentally qualified. In addition, the Reactor Building Purge Exhaust System will not be shut down on actuation (opening) of the deluge valve, since this component is not environmentally qualified either. However, actuation of the alarms associated with the above mentioned components, although not qualified to function, will not be prevented from functioning due to the modification.

   The above mentioned modification should increase the reliability of the control system, which should ensure the operation of the Reactor Building Purge Exhaust System (Hydrogen Purge) under Post-LOCA conditions.

2. No, the 3-way hand valves on AH Panel 13 and the key switches installed in the Motor Control Centers (MCC), for AHF-7A and AHF-7B, are Administratively controlled.

   The 3-way hand valves are only used to bypass the non-environmentally qualified solenoid valves, which are used to position (open and close) the dampers associated with the fans. No electrical permissives or shutdown functions are bypassed when only the 3-way hand valves are operated. Under normal operation of the Reactor Building Purge Exhaust System, the dampers must be fully open to allow the fans to start. Thus, if the 3-way hand valves are mis-positioned to the open position (Emergency position) during normal operation of the system, the fans will start, if selected to start. Additionally, non-environmentally qualified limit switches provide

MAR 90-08-014-01A
Reactor Building Purge Exhaust System
(Continued)

indication of the dampers position to the control room, for normal operating conditions.

The key for the key switches, which provide the method for bypassing the non-environmentally qualified equipment, can only be removed in the "Normal" position. Since the bypass capability will only be enabled for Post-LOCA, sufficient administrative controls exist to prevent the possibility of the switches being put in the "Emergency" position inadvertently.

The use of the damper indication, via limit switch indication in the control room and visual field inspection, will be used to Administratively start the fans during Post-LOCA operation. Per FSAR Chapter 14B, purging is not assumed to begin prior to 10 days Post-LOCA. In addition, a concentration of 3.5% hydrogen is reached 29.5 days Post-LOCA. Purging would begin in 25 days (600 hours), which would result in an operators dose of less than 200 mrem to manually operate dampers AHD-94 through AHD-98 via the 3-way hand valves. The above mentioned dose to operate the 3-way valves following a postulated LOCA can be accomplished without exceeding the operator dose limit of 5 Rem, which is based on the dose requirements established in 10 CFR 50 Appendix A, GDC 19 as outlined in NUREG-0737.

The above mentioned modification should increase the reliability of the control system, which should ensure the operation of the Reactor Building Purge Exhaust System (Hydrogen Purge) under Post-LOCA conditions.

3. No, former Technical Specification 3.6.4.2 stated: "A containment hydrogen purge system shall be OPERABLE." The Reactor Building Purge Exhaust System provides a path for the containment hydrogen purge system through LRV-70/71 and LRV-72/73 to AHF-7A and AHF-7B and finally through the plant stack. The System remains operable with this change.

This modification was initiated to address the concerns of the Non-EQ electrical equipment associated with the control of the dampers (AHD-94, AHD-95, AHD-96, ADD-97 and AHD-98) and the Air Handling Fans (AHF-7A and AHF-7B), as being permanently installed.

The modification should increase the reliability of the Reactor Building Purge Exhaust System (Hydrogen Purge) by bypassing (defeating) the electrical controls for the dampers and permissive start/run controls of the fans, since the existing controls do not meet the EQ requirements for operating Post-LOCA.

MAR 93-12-07-02
Reactor Building Penetrations

1. No, the DBA occurrence or frequency change will not result from the design change of theses RB penetrations. Containment pressure boundary design basis is maintained by virtue of the design, procurement, fabrication, inspection, and test standards invoked.

2. No, these new penetration designs are subject to the same construction standards and leak testing applied to others discussed in the FSAR. Containment pressure boundary performance is not compromised, therefore accident consequences are not increased.

3. No, new failure modes or accident scenarios are not created as a result of these 2 penetration design changes. Penetration design criteria is well established. All original standards applied to the penetrations are duplicated, therefore no new conditions are present.

4. No, allowable leakage criteria is not changed for the containment structure. The equipment malfunctions discussed in the FSAR are not affected by this change, and neither is the Safety Margin of containment of a pressure boundary or barrier to the public.

5. No, since the penetration's design, fabrication, inspection and test standards duplicate those for existing designs, the consequences of a DBA are unchanged. The double barrier minimizes the potential of leakage through the new seals (gaskets) irrespective of the maintenance and testing standards applied.

6. No, a new credible failure condition is not created by the modification of these penetrations. The function of the pressure boundary requirement (passive) is not altered. The total allowable leakage is not altered. This penetration design type is already discussed in the FSAR. Ref: 5.6.4.2 and Table 5-4.

7. No, the design and construction standards of these penetrations maintain the margin of safety for all operating modes. During outages (modes 5/6), the blind flange assembly design provided meets the intent of Section 3.9.4. Accordingly, through pre-operational and periodic testing, containment integrity is maintained under all conditions required.

MAR 93-08-09-01
Incore Monitors

This MAR has been initiated to replace the existing "Short" Rh emitter incore detector assemblies with "Long" Rh emitter incore detector assemblies. The existing "Short" Rh emitter detector assemblies contain 4.75 inch Rh emitters. The new "Long" Rh emitter detector assemblies contain 15.75 inch Rh emitters.

The thermocouples and background detectors are of the same design in the new and existing incore detector assemblies. Only the Rh emitters are being lengthened in the new detector assemblies, which will increase the life expectancy of the emitters/incore detector assemblies. In addition, no exterior physical changes have been made; the new and existing assemblies meet the same design requirements.

1. No, this modification will replace the existing "Short" Rh emitter incore detector assemblies with "Long" Rh emitter incore detector assemblies.

   The IM System provides no protective or control functions. The Rh emitters are not required to function during and after a Design Basis Event (DBE). The only portion of the incore detector assemblies that are required to function during and after a DBE are the 16 Class 1E thermocouples. Since the Class 1E and Non-Class 1E thermocouples are of the same design for the "Short" and "Long" SPNDs; the function of the thermocouples for Post-DBE operation as initially designed will not be affected.

2. No, per Section 4.1.2 of Enhanced Design Basis Document for the Nuclear Instrumentation System, the Safety Requirements of the Incore Monitoring System are:

   "The incore monitoring system must provide recording of 16 core exit temperatures in the control room, four in each quadrant for the inadequate core cooling system."

   "The incore monitoring system must provide closure of the reactor coolant system boundary."

   Since neither the Class 1E (nor Non-Class 1E) thermocouples nor the exterior construction of the SPNDs are changed between the "Short" Rh SPNDs and the "Long" Rh SPNDs, the above safety functions will not be affected by the installation of this modification.

   FSAR Section 7.3.3.2.1 discusses the sixteen (16) Class 1E thermocouples. The Class 1E and Non-Class 1E thermocouples are not being modified/affected by this modification. The Class 1E and Non-Class 1E thermocouples are the same in the "Short" and the "Long" incore detector assemblies.

MAR 93-08-09-01
Incore Monitors
(Continued)

3. No, the Rhodium (Rh) emitters associated with the Incore Detector Assemblies are not required to function during or Post-DBE (Design Basis Event); however, the detector assemblies are Safety Related from a pressure boundary standpoint.

   The design of the Class 1E and Non-Class 1E thermocouples is not being affected by this modification. In addition the design of the pressure boundary (outer sheath) of the detector assembly is not being affected by the lengthening of the Rh detectors. Therefore, the Safety functions/components of the detector assemblies are not being affected.

4. No, presently B&W updates the Nuclear Application Software (NAS) every refuel to account for incore detector burn-up to ensure that the current output of the emitters reflects the correct reactor power level. B&W will update the NAS during Refuel 9 and subsequent refuels, to ensure that the computer will correlate the current output from the "Long" emitters to the correct reactor power level.

   The only difference between the "Short" and "Long" emitters is the length of the Rh detectors, which affects the current signal to the plant computer. A mixture of "Long" and "Short" Rh emitters detectors will exist until the completion of this Generic MAR is complete; when all "Short Rh detectors have been replaced with "Long" Rh detectors. The NAS will be updated by B&W Fuel Company (BWFC) and/or the computer specialist, as required, to correlate the correct signal input to reactor power level to ensure the correct operation of the Incore Detector System.

5. No, all of the components in the "Long" incore detector assemblies are identical to the "Short" incore detector assemblies, except for the length of the Rh emitters. The Rh emitter are used to determine reactor power, but due to the delay time of the Rh emitters, the output from the Rh emitters are not used for any automatic/control functions.

6. No, the Normal current range of the "Short" Rh emitter incore detector assembly is 0 to 1000 nA, with a Maximum range of 2000 nA. The Normal current range of the "Long" Rh emitter incore detector assembly is 0 to 3000 nA, with Maximum range of 6000 nA. Since the Maximum range is twice as large as the Normal range, this ensures that any credible localized power excursions will be within the range of the emitter.

7. No, the only change between the "Short" and "Long" SPNDs is the change to the current output. Even though the signal output will increase, this will not result in any Technical Specification changes, because the Nuclear Application Software (NAS) will be updated to correlate signal input to reactor power. Therefore, no parameters which would affect Technical Specification are being affected.

MAR 93-06-16-01
Nuclear Instrumentation

This MAR provides three functional changes to CR-3 which are as follows:

A) Provide MCB indication of Source Range Count Rate and Source Range Rate of Count Rate Change such that they can be compared with the existing Bailey Source Range Indications in order for the operators to become acquainted with the performance of Fission Chambers for Source Range monitoring.

B) Provide Appendix R Fire protection to circuits of NI-14 in the RB and IB and to a NI-15 circuit in the "B" ES 480 VAC SWGR room in order to take credit for NI-14 and NI-15 as backup Indication for the Technical Specification Requirement for two Channels of Source Range Indication being available for Reactor Startup. This backup capability does not include the Rod Withdrawal Inhibit or Alarm functions like the NI-1 & 2 channels, therefore when utilized as such, procedures alerting the operators that these functions are to be provided manually must be implemented. During Physics Tests, the current Technical Specifications on Special Test Exceptions require the Rod Withdrawal Inhibit Function. This prohibits use of NI-14 and NI-15 as backups unless modified to provide the interlock. The new Improved Technical Specifications do not require this function be in place during physics testing. In addition, Technical Specification Surveillance requirements are also required on NI-14 and NI-15 when they are used to satisfy the Technical Specification requirement for a normal startup.

C) Change Gamma-Metrics Instrumentation (NI-14 & 15) to improve Instrument Accuracy and change the Reactor Power Level Set-Point for AMSAC Enable from 25% to 45%. This higher Enable Set-Point is justified by analysis which demonstrated that a ATWS Event at a true 50% Reactor Power would not have unacceptable consequences. The improved Instrument accuracies reduced the string error in the Set-Point determination allowing the setpoint increase to be closer to 50%.

1. No, each functional change of the MAR is addressed below:

    A) The added MCB Indication like the existing Indication provides no mechanism for accident initiation and as such can not increase probability for occurrence.

    B) The Source Range High Start-Up Rate Rod Withdrawal Inhibit and Alarm are non-safety related functions that are not taken credit for in the FSARs Startup Accident analysis, however, they are identified as one of the system features that minimize the possibility of inadvertent continuous rod withdrawal. The absence of these functions while in the Source Range could provide a minor increase in event occurrence probability. This minor increase in probability is offset by procedural controls stressing the requirement for Manual Startup

MAR 93-06-16-01
Nuclear Instrumentation
(Continued)

Rate Limitation when operating in this mode. This approach is acceptable under NSAC-125 guidelines which states: "Compensating effects such as changes in administrative controls may be used to offset an increase or trend in the probability of accidents of moderate frequency".

C) The increase in AMSAC Enable Set-Point reduces the potential for plant upsets while at power levels below which AMSAC is beneficial, which in turn reduces the potential for a spurious AMSAC initiated transient.

2. No, each functional change of the MAR is addressed below:

A) The additional Indication provides no increase in accident consequences because no credit has been taken for the existing Source Range Indication in accident mitigation.

B) Use of the backup Source Range Indication without a Rod Withdrawal Inhibit Function would not increase the consequences of an accident because the existing Indication with a Rod Withdrawal Inhibit Function was not taken credit for in accident mitigation.

C) The higher AMSAC Enable setpoint is justified by analysis which demonstrated that an ATWS Event at a true 50% reactor power results in a peak RCS pressure of 3005 psig which is an acceptable ATWS Peak per the B&W Owners Group Design Criteria. The existing 25% Enable setpoint was not analyzed to determine a resulting peak pressure because the EFW System with existing Safety Valve Capacity is capable of removing 25% to 35% Reactor Power and, as such, would not result in a peak pressure significantly above normal. The predicted peak with the 50% enable setpoint is above that expected with the existing 25% setpoint, however, this does not increase the consequences for the general public because it is still below the B&W Owners Group ATWS limits which were set considering that Reactor Fuel, RC System, or Containment boundaries would not be breached. The actual new setpoint includes instrument error to assure true reactor power is not over 50% without AMSAC Mitigation in effect.

3. No, each functional change of the MAR is addressed below:

A) The added MCB Indication like the existing Indication provides no mechanism for accident initiation and as such would not cause an accident of different type.

MAR 93-06-16-01
Nuclear Instrumentation
(Continued)

B) Use of the backup Source Range Indication without a Rod Withdrawal Inhibit Function is like the existing Indication that provided no mechanism for accident initiation and as such would not cause a accident of different type.

C) The increase in AMSAC Enable Set-Point reduces the potential for plant upsets while at power levels below which AMSAC is beneficial, which also reduces the potential for a spurious AMSAC initiated transient which could be of a different type.

4. No, each functional changes of the MAR is addressed below:

A), B), C) The added Instrumentation Indication and AMSAC outputs are 1E isolated such that failures can not propagate back to the Regulatory Guide 1.97 Part of the NI-14 & 15 Circuits and the modifications to the Gamma-Metrics Equipment was purchased to maintain it's original qualifications. Therefore, this modification would not increase the probability of critical equipment malfunction.

5. No, each Functional Change of the MAR is addressed below:

A) The new Dual Pin Indicators have the same failure mode as the existing Single Pin Indicators. Therefore, with the same failure mode the consequences do not change.

B) This modification does not change the Mode of failure of existing instrumentation. Therefore, with the same failure mode the consequences do not change.

C) The increased power level where an ATWS event could occur without AMSAC available has a higher peak pressure for the extremely unlikely case of no DSS actuation but it is within the original B&W Owners Group Criteria such that the consequences to the general public are not increased.

6. No, each functional change of the MAR is addressed below:

A) The new Dual Pin Indicators have the same failure mode as the existing Single Pin Indicators, thus failure effects would not be different and as such does not cause additional equipment failure possibilities.

MAR 93-06-16-01
Nuclear Instrumentation
(Continued)

B) & C) This modification does not change existing interface equipment and as such does not cause additional equipment failure possibilities.

7. No, each functional change of the MAR is addressed below:

A) & B) The existing indication is addressed by Technical Specification as a condition for mode change and for a special test exception (Physics Tests). This MAR provides the capability to use two additional Instrument strings for this purpose except for the Physics Testing condition. The capability to utilize a Backup Source Range Indication for Plant Startup does not reduce any safety margin because it is Operator information and Operator action like the Rod Withdrawal Inhibit function is not relied upon to provide termination of the Reactivity addition. The High Flux or overpressure Trips provide termination of the Reactivity addition by a Reactor shutdown.

C) ATWS Mitigation is not addressed by Technical Specifications, however, there is a special commitment to the NRC for the capability to maintain RCS Peak pressure from a ATWS event below the B&W Owners Group ATWS limits. The acceptability of new setpoint is demonstrated by analysis.

Boron Dilution For LOCA Mitigation

1. No, accrediting the reactor vessel internal gap flow path as a passive method of post-LOCA boron dilution as replacement for the decay heat drop line and the hot leg injection dilution method will not increase the probability of occurrence of an accident previously evaluated in the FSAR because neither flow path is an accident initiator. The gaps exist due to the reactor vessel design and construction. The size of the gaps varies depending on actual construction and relative temperatures in the various parts of the CR-3 reactor vessel. Because the presence of the gaps was included in the design of the reactor vessel, their effects have been inherently included in the FSAR analyses. The gaps do not create any adverse conditions which could initiate any of the accidents previously evaluated in the FSAR. Therefore, the probability of occurrence has not changed.

2. No, accrediting the reactor vessel internal gap flow path as a passive method of post-LOCA boron dilution as replacement for the decay heat drop line and the hot leg injection dilution methods will not increase the radiological consequences of an accident previously evaluated in the FSAR since it has been determined that adequate dilution flow will still exist to prevent boric acid precipitation in the reactor vessel. This ensures the core coolant channels will not be blocked by precipitate so that core cooling is assured. This limits the fission product release from the core to that previously analyzed. Therefore, the radiological consequences of all accidents presented in the FSAR are unchanged.

3. No, accrediting the reactor vessel internal gap flow path as a passive method of post-LOCA boron dilution as replacement for the decay heat drop line and the hot leg injection dilution methods will not increase the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the FSAR because the flow through the gaps does not adversely affect any equipment important to safety. The gaps have always been present in the reactor vessel. Their presence has no adverse affect on any equipment important to safety. The reactor vessel design included the gaps, so there is no change in the vessel's capability. The presence of the gaps does not affect the ability to keep the core cooled under any conditions. Initiation of the drop line dilution path is not affected by the gaps. Since the gap flow path is a passive mechanism which does not require any additional liquid, beyond that previously analyzed, to be injected into the Reactor Coolant System, there is no change in the containment's environmental response to a LOCA. Deleting a requirement for either the decay heat drop line or the hot leg injection flow path to be operable does not increase the probability of equipment malfunctioning. The probability of the decay heat drop line or the auxiliary spray line failing to provide dilution flow following a large break LOCA has been eliminated by replacing those functions with the internal gap flow dilution method. Consequently, for the dilution function, the decay heat drop line and the auxiliary spray line is not considered equipment important to safety and do not enter into evaluating

Boron Dilution For LOCA Mitigation
(Continued)

the probability that equipment important to safety will malfunction. Therefore, the probability of equipment important to safety malfunctioning has not changed.

4. No, accrediting the reactor vessel internal gap flow path as a passive method of post-LOCA boron dilution as replacement for the decay heat drop line and the hot leg injection dilution methods will not increase the radiological consequences of a malfunction of equipment important to safety previously evaluated in the FSAR because analyses have shown that the reactor vessel gaps will permit boron dilution such that core damage does not occur which exceeds that previously evaluated in the FSAR. The flow through the gaps does not contribute any potential for increased radioactive release from the containment. The boron concentration in the core has been analyzed to stay below the solubility limit so no additional core radioactivity release will occur. Therefore, there is no increase in the radiological consequences of a malfunction of equipment important to safety previously evaluated in the FSAR.

5. No, accrediting the reactor vessel internal gap flow path as a passive method of post-LOCA boron dilution as replacement for decay heat drop line and the hot leg injection dilution methods will not create a possibility for an accident of a different type than previously evaluated in the FSAR because the gaps have always existed and have been included in the vessel design. Analysis has shown that boron precipitation will not occur, so that no different type of plant response to a LOCA can occur due to the gaps. Therefore, no new accident scenarios can be postulated, so there is no possibility for creating an accident of a different type.

6. No, accrediting the reactor vessel internal gap flow path as a passive method of post-LOCA boron dilution as replacement for the decay heat drop line and the hot leg injection dilution methods will not create the possibility for a malfunction of equipment important to safety of a different type than any previously evaluated in the FSAR because the method does not interact with any other equipment important to safety. The presence of the gaps has already been included in the design of the reactor vessel so that no new failure mechanism of the vessel need be postulated.

7. No, accrediting the reactor vessel internal gap flow path as a passive method of post-LOCA boron dilution as replacement for the decay heat drop line and the hot leg injection dilution methods will not reduce the margin of safety as defined in the basis for any Technical Specification because analysis has shown that boron precipitation will not occur, so that the post-LOCA response will not change. The Technical Specifications and their bases define the equipment and plant operating requirements necessary to ensure the plant responds to design bases events as analyzed in the FSAR. Because the passive gap flow path will be available

Boron Dilution For LOCA Mitigation
(Continued)

immediately following a LOCA and provides sufficient dilution without relying on active systems or operator action, no specific dilution initiation time requirement needs to be specified to preserve the margin of safety as defined in the Technical Specifications. Therefore, the margin of safety is not reduced.

FSAR 12.8.1.6(a)
Administrative Controls

This change to Section 12.8.1.6(a) of the FSAR will correct the required Director, Nuclear Plant Operations (DNPO) approvals for activities reviewed by the Plant Review Committee (PRC) that were inaccurately relocated during the Improved Technical Specification implementation at Crystal River Unit 3. Several activities were inadvertently omitted from DNPO approval and one activity was added by mistake. This revision will place changes to the Fire Plan and implementing procedures, the Process Control Program, and the Offsite Dose Calculation Manual under the approval of the DNPO prior to implementation as originally required in CR-3's Technical Specifications. The requirement for the DNPO to approve all procedures and changes thereto is being deleted since this is not consistent with the original requirements of the Technical Specifications and represents an increased administrative burden for the DNPO with no added benefit to safety. These activities were not properly relocated to the FSAR as a result of an oversight during the transition activities associated with the implementation of the Improved Technical Specifications.

1. No, the administrative requirement for DNPO approval of the above listed items does not affect the probability of occurrence of accidents. The procedures and programs listed received and will continue to receive proper inter- and intra-departmental reviews and manager approvals prior to implementation. These reviews and approvals are adequate to maintain control over proceduralized activities for safe operation of the unit.

2. No, the use of DNPO approval on these activities does not assure improved procedural guidance or worker adherence to the requirements established in the documents. Therefore, this change will not cause conditions that would degrade system performance or personnel responses to design basis accidents.

3. No, the departmental reviews and approvals of these procedures and programs assures adequate consideration has been given to conditions which could be outside the bounds of existing evaluated FSAR design basis accidents. The requirement for the DNPO approval does not significantly add or detract from these controls.

4. No, this change re-establishes the original requirements for DNPO approval of these activities. Reviews and approvals by affected departments remain adequate to control functions which are necessary for the safe operation of the unit.

5. No, the administrative approval process for the DNPO does not adversely affect the primary activities for the operation and maintenance of equipment necessary for the safe operation of the unit. Therefore, the consequences of equipment malfunctions on design basis events described in the FSAR are unaffected by this change.

FSAR 12.8.1.6(a)
Administrative Controls
(Continued)

6. No, this change corrects the administrative controls originally required by Technical Specifications, and, therefore, no increase in the possibility of equipment malfunction is created.

7. No, this change corrects the approval requirements for the DNPO associated with PRC reviewed items. The relocation of these requirements to the FSAR during implementation of ITS was inconsistent with the original requirements of Tecnnical Specifications. The attached change corrects the FSAR to be consistent with the original statements of DNPO approval activities. Since this activity was relocated to the FSAR and this change is to make the relocated statements consistent with the original Technical Specification requirements, no reduction in the margin of safety as defined in the Technical Specification bases is created.

MAR 88-05-25-05
Nuclear Service Closed Cycle Cooling (SW) System
Isolation to the RC Pumps and Reactor Building Coolers

1. No, the modification changes the logic for the cooling water isolation valves, SWV-79, SWV-80, SWV-81, SWV-82, SWV-83, SWV-84, SWV-85, and SWV-86, to close the valves when the Nuclear Services Closed Cycle Cooling Water (SW) System Surge Tank, SWT-1, level reaches a predetermined setpoint coincidental with an ES Actuation on high Reactor Building (RB) Pressure. This change permits the SW system lines to the Reactor Coolant (RC) Pumps to remain open for scenarios where RB Pressure is high (greater than 4 PSI) and the SW system is intact. With this change in-place, operator burden is reduced under certain Design Basis Accidents, since the RC Pumps can remain operating (assuming power is available) without concern for potential pump damage. The SW lines to the RC Pumps are normally open supplying cooling water to the RC pumps. The normally open condition is unchanged by this modification since the logic change will only take affect coincident with an ES Actuation. The level logic is coincident with the ES Actuation, therefore, the potential for spurious operation, of the subject SW system valves, causing a loss of SW to the RC Pumps is reduced since both conditions must occur to cause isolation.

   The modification to the logic for SW to the Reactor Building Cooling Unit (RBCU) reduces the heat load on the SW system by assuring that water flows only through the RBCUs which are operating. The logic to open the valve is basically unchanged, since when starting a fan on slow speed the associated cooling water isolation valve control circuit will be de-energized and the valve will open to its normal position. However, unlike the previous design which required operator action to close the valve, de-energizing the RBCU fan control circuit will result in the automatic closure of the cooling water return (discharge) line's valve (SWV-41,SWV-43, or SWV-45). A time delay prevents unnecessary cycling of the valve during fan speed changes, a Loss of Offsite Power (LOOP), or ES Actuation (Diesel Start and Block Loading). This change will reduce operator burden and, at the same time, provide a higher flow capacity for other ES Components served by the SW system.

   The failure or malfunction of any equipment associated with this modification would not contribute to initiating the accidents identified by FSAR Sections 14.1.2.9, 14.2.2.1, 14.2.2.5.6, and 14.2.2.5.7.

   With respect to the Station Blackout Accident, the isolation of cooling water to the RC Pumps and/or the RBCUs will not increase the probability of occurrence of this accident as previously evaluated in FSAR Section 14.1.2.9.

MAR 88-05-25-05
Nuclear Service Closed Cycle Cooling (SW) System
Isolation to the RC Pumps and Reactor Building Coolers
(Continued)

With respect to the S!_OCA, the cooling water to the RC Pumps is provided by the SW System during both normal and emergency plant conditions. Cooling water to the RBCUs is provided by the industrial coolers during normal plant conditions and by the SW system during emergency plant conditions. The SW System is not part of the RCS pressure boundary. Therefore, the isolation of cooling water to the RC Pumps and/or the RBCUs will not increase the probability of the occurrence of this accident as previously evaluated in FSAR Section 14.2.2.5.7.

An analysis has been performed which assumed the failure of a 3" SW line to RCP Seal Area Cooler. That flow did not exceed 3820 gpm. FPC has determined that this flow rate from the surge tank will not result in any loss of SW system operability. In addition this modification will isolate SW flow from the RBCU which is not operational following the four pound isolation signal. Addition of the RCP cooling during an accident situation and deletion of an inoperative RBCU resulted in:

1) a decrease in required SW system flow post ES actuation

2) an increase in SW system heat load post ES actuation

3) a decrease in Emergency Diesel Generator loading post ES actuation.

The new post accident flow rate required from the SW system will be 8071 GPM. A rebalance of the SW system will take place to assure all the components which receive SW flow post ES will receive their required flow. The increase in SW heat lcad will be acceptable because the new calculated peak temperature, with 95F RW, will be 108.9F. All SW cooled components which receive flow post ES are shown to be able to perform their required functions with SW cooling water up to 110F. The decrease in Emergency Diesel Generator load, as a result of the decreased SW flow will have no negative effect on plant safety.

2. No, the only potential for increasing the consequences of an accident previously evaluated is if the release or potential for release to the environment was increased. The subject SW system isolation valves are all normally open, energize to close, and are provided with DC powered control circuits. The potential release is not increased since the SW system is a closed loop inside the RB. This system continues to satisfy GDC 54 and 57. This modification does not charge the containment isolation functions of the SW system penetrations which serve the RBCUS.

MAR 88-05-25-05
Nuclear Service Closed Cycle Cooling (SW) System
Isolation to the RC Pumps and Reactor Building Coolers
(Continued)

The consequences of the Station Blackout Accidents, in terms of dose to the public, are related to high RCS pressures and high secondary side pressures. The isolation of cooling water to the RC Pumps and/or the RBCUs will not increase the consequences of this accident previously evaluated in FSAR Section 14.1.2.9.

The consequences of the SBLOCA, in terms of dose to the public, are related to the location and amount of leakage from the RCS and containment isolation. The SBLOCA could be a condition where it is desirable to continue operating the RC Pumps, and thus, require continuous operation of the SW System cooling water to the RC Pumps. This modification adds instrumentation and interlocks to the containment isolation valves, in the RC Pumps cooling water lines, to prevent automatic closure of these valves upon ES Actuation (Reactor Building Isolation and Cooling) unless it is coincident with the sensed level setpoint in SW system surge tank that may be indicative of an SW System line break. The modification does not affect the ability to manually close the RC Pump cooling water isolation valves from the ESF section of the Main Control Board (MCB) in the Main Control Room (MCR).

Also, the modification does not change the control logic of the valves in the cooling water lines to the RBCUS, which is designed to prevent these valves from closing (isolating coolant) to a running RBCU.

3. No, the equipment used for this modification is considered to be Nuclear Safety Related. This modification interfaces with SW System components and electrical circuits that are Safety Related, thus, the work to be performed is Safety Related.

4. No, the SW System subject valve(s) are all normally open and will fail open upon loss of control circuit DC Power or loss of air. The modification will not have any affect on the valve(s) basic safe position.

5. No,this modification does not create any new accident scenarios. The SW System valves have been designed to perform the safety function to provide cooling water to the RC pumps and RBCU'S, and to meet the criteria for Reactor Building Isolation In that the SW system lines to the RC Pumps and the RCBUs are closed loops inside the Reactor Building, the GDC for Reactor Building isolation are still satisfied. The failure mode on loss of power or air is not changed by this modification.

MAR 88-05-25-05
Nuclear Service Closed Cycle Cooling (SW) System
Isolation to the RC Pumps and Reactor Building Coolers
(Continued)

6. No, the addition of a time delay relay, to the control circuit of each RBCU cooling water discharge valve has been designed as Safety Related. The only failure which could result from this modification is the failure of the automatic closure of the valve when the associated fan is not running. The control circuit for each valve remains independent. The addition of the time delay will not prevent the valve from opening when the associated fan is started, or prevent the valve from being manually closed from the MCR if the fan is not running and if the time delay relay fails to automatically close the valve.

   The addition of an interlock in the control circuit of the containment isolation valves to the RC Pumps will not cause the other equipment in the respective circuit to malfunction. Since the SW system lines are allowed to remain open for more accident scenarios, the concern would be an event which could increase site boundary exposure. The closed SW system loop inside the Reactor Building will prevent fission product release. Each of these valves has redundant control circuits powered from Train A and Train B DC Power supplies. This modification will install redundant SW system instrumentation to interlock with the redundant control circuit for each valve.

7. No, this modification does not affect any of the Technical Specification parameters, there is no effect on the margin of safety. This modification will have no effect on the operation of any post-ES actuation SW cooled components. Deletion of SW cooling water to the non-operating RBCU will reduce the post-ES actuation load on the SW system and will reduce the load on the emergency diesel generators. The SW system will be rebalanced in order to assure that all components receive their required flow.

FSAR Section 12.8.1.6(a) Change

This change to FSAR Section 12.8.1.6(a) will correct the required Director, Nuclear Plant Operations (DNPO) approvals for activities reviewed by the Plant Review Committee (PRC) that were inaccurately relocated during the Improved Technical Specification implementation at Crystal River Unit 3. Several activities were inadvertently omitted from DNPO approval and one activity was added by mistake. This revision will place changes to the Fire Plan and implementing procedures, the Process Control Program, and the Offsite Dose Calculation Manual under the approval of the DNPO prior to implementation as originally required in CR-3's Technical Specifications. The requirement for the DNPO to approve all procedures and changes thereto is being deleted since this is not consistent with the original requirements of the Technical Specifications and represents an increased administrative burden for the DNPO with no added benefit to safety. These activities were not properly relocated to the FSAR as a result of an oversight during the transition activities associated with the implementation of the Improved Technical Specifications.

1. No, the administrative requirement for DNPO approval of the above listed items does not affect the probability of occurrence of accidents. The procedures and programs listed received and will continue to receive proper inter-departmental and intra-departmental reviews and manager approvals prior to implementation. These reviews and approvals are adequate to maintain control over proceduralized activities for safe operation of the unit.

2. No, the use of DNPO approval on these activities does not assure improved procedural guidance or worker adherence to the requirements established in the documents. Therefore, this change will not cause conditions that would degrade system performance or personnel responses to design basis accidents.

3. No, the departmental reviews and approvals of these procedures and programs assures adequate consideration has been given to conditions which could be outside the bounds of existing evaluated FSAR design basis accidents. The requirement for the DNPO approval does not significantly add or detract from these controls.

4. No, this change re-establishes the original requirements for DNPO approval of these activities. Reviews and approvals by affected departments remain adequate to control functions which are necessary for the safe operation of the unit.

5. No, the administrative approval process for the DNPO does not adversely affect the primary activities for the operation and maintenance of equipment necessary for the safe operation of the unit. Therefore, the consequences of equipment malfunctions on design basis events described in the FSAR are unaffected by this change.

FSAR Section 12.8.1.6(a) Change
(Continued)

6. No, this change corrects the administrative controls originally required by Technical Specifications and, therefore no increase in the possibility of equipment malfunction is created.

7. No, this change corrects the approval requirements for the DNPO associated with PRC reviewed items. The relocation of these requirements to the FSAR during implementation of ITS was inconsistent with the original requirements of Technical Specifications. The attached change corrects the FSAR to be consistent with the original statements of DNPO approval activities. Since this activity was relocated to the FSAR and this change is to make the relocated statements consistent with the original Technical Specification requirements, no reduction in the margin of safety as defined in theTechnical Specification bases is created.