WHOLE BODY EXPOSURE FOR CALENDAR YEAR 1992

Licensee Reporting (Name and Address)	NRC license No(s).
FLORIDA POWER CORPORATION CRYSTAL RIVER UNIT 3 P. O. BOX 219 CRYSTAL RIVER, FLORIDA 32623-0219	DPR-72

| | If personnel monitoring was not required to be provided to any individual during the year, checking this box will constitute a negative report indicating that such personnel monitoring was not required.

OTHERWISE, COMPLETE THE FOLLOWING TABLE:

Estimated Annual Whole Body Exposure Ranges * (Rems)	Number of Individuals in Each Range
No Measurable Exposure	1025
Less than 0.100	538
0.100 - 0.250	296
0.250 - 0.500	276
0.500 - 0.750	139
0.750 - 1.000	70
1.000 - 2.000	82
2.000 - 3.000	2
3.000 - 4.000	0
4.000 - 5.000	0
5.000 - 6.000	0
6.000 - 7.000	0
7.000 - 8.000	0
9.000 - 9.000	0
9.000 - 10.000	0
10.000 - 11.000	0
11.000 - 12.000	0
Greater than 12.000	0
TOTAL NUMBER OF INDIVIDUALS REPORTED	2428

The above information is submitted for the total number of individuals for whom personnel monitoring was: (check one)

| | Required under 10 CFR 20.202(a) or 10 CFR 34.33(a) during the calendar year.

|X| Provided during the calendar year.

* Individual values exactly equal to the values separating exposure ranges shall be reported in the higher range.

FLORIDA POWER CORPORATION
ANNUAL STS DOSE REPORT
FOR PERIOD 01/01/1992 TO 12/31/1992

	NUMBER OF PERSONNEL (>= 100 mREM)			TOTAL MAN-REM		
	STATION EMPLOYEES	UTILITY EMPLOYEES	CONTRACT WORKER AND OTHERS	STATION EMPLOYEES	UTILITY EMPLOYEES	CONTRACT WORKER AND OTHERS
REACTOR OPERATIONS AND SURVEILLANCE						
MAINTENANCE	0	0	0	0.012	0.074	0.012
OPERATION PERSONNEL	26	0	0	7.007	0.005	0.000
HEALTH PHYSICS PERSONNEL	0	0	0	0.000	0.000	0.000
SUPERVISOR PERSONNEL	0	0	0	0.000	0.032	0.005
ENGINEERING PERSONNEL	0	0	0	0.000	0.100	0.000
ROUTINE MAINTENANCE						
MAINTENANCE	100	35	254	37.402	11.846	134.977
OPERATION PERSONNEL	8	4	65	2.498	1.059	25.275
HEALTH PHYSICS PERSONNEL	48	1	67	14.618	1.010	31.537
SUPERVISOR PERSONNEL	0	0	146	0.147	0.574	74.435
ENGINEERING PERSONNEL	0	4	11	0.000	2.069	5.148
INSERVICE INSPECTION						
MAINTENANCE	0	0	40	0.014	0.016	16.928
OPERATION PERSONNEL	0	0	9	0.000	0.000	2.343
HEALTH PHYSICS PERSONNEL	0	0	8	0.111	0.000	2.638
SUPERVISOR PERSONNEL	0	0	26	0.000	0.000	8.720
ENGINEERING PERSONNEL	0	0	1	0.000	0.089	0.208
SPECIAL MAINTENANCE						
MAINTENANCE	0	0	0	0.000	0.000	0.000
OPERATION PERSONNEL	0	0	0	0.000	0.000	0.000
HEALTH PHYSICS PERSONNEL	0	0	0	0.000	0.000	0.000
SUPERVISOR PERSONNEL	0	0	0	0.000	0.000	0.000
ENGINEERING PERSONNEL	0	0	0	0.000	0.000	0.000
WASTE PROCESSING						
MAINTENANCE	8	0	0	2.116	0.000	0.064
OPERATION PERSONNEL	1	0	1	0.245	0.000	0.251
HEALTH PHYSICS PERSONNEL	15	0	0	8.344	0.000	0.166
SUPERVISOR PERSONNEL	0	0	0	0.000	0.000	0.003
ENGINEERING PERSONNEL	0	0	0	0.000	0.000	0.000
REFUELING						
MAINTENANCE	0	0	34	0.515	0.235	18.735
OPERATION PERSONNEL	0	0	11	0.084	0.038	2.387
HEALTH PHYSICS PERSONNEL	1	0	6	0.379	0.000	1.376
SUPERVISOR PERSONNEL	0	0	16	0.000	0.007	8.543
ENGINEERING PERSONNEL	0	0	0	0.000	0.052	0.000
TOTALS						
MAINTENANCE	108	35	328	40.059	12.171	170.716
OPERATION PERSONNEL	35	4	86	9.834	1.102	30.256
HEALTH PHYSICS PERSONNEL	64	1	81	23.452	1.010	35.717
SUPERVISOR PERSONNEL	0	0	188	0.147	0.613	91.706
ENGINEERING PERSONNEL	0	4	12	0.000	2.310	5.356
			TOTAL DOSE IN REMS BY CATAGORY	73.492	17.206	333.751
			TOTAL STATION DOSE IN REMS	424.449		

ANNUAL REPORT OF REACTOR VESSEL MATERIAL SURVEILLANCE CAPSULES AT CRYSTAL RIVER UNIT 3 IN ACCORDANCE WITH TECHNICAL SPECIFICATION 6.9.1.5(b)

REPORTING PERIOD
JANUARY 1, 1992 TO DECEMBER 31, 1992

CAPSULES CURRENTLY INSTALLED IN THE REACTOR

Holder Tube	Position In Holder Tube	Capsules Installed
ZY	Top	OC3-F
	Bottom	OC1-D
XW	Top	Empty
	Bottom	Empty
YX	Top	A2
	Bottom	A4
YZ	Top	TMI2-LG1
	Bottom	TMI2-LG2
WZ	Top	OC3-E
	Bottom	CR3-LG2
WX	Top	OC2-F
	Bottom	TMI1-D

INSERTION/WITHDRAWAL SUMMARY

End of Eighth Fuel Cycle

Holder Tube	Position in Holder Tube	Capsule Withdrawn	Capsule Inserted	Capsules Installed
ZY	Bottom	CR3-A	OC1-D	2
XW	Top	CR3-A	None	2
XW	Bottom	TMI-2D	None	2
WX	Top	OC3-C	OC2-F	2
WX	Bottom	TMI1-F	TMI1-D	2

OWNERSHIP OF CAPSULES

Florida Power Corporation

No capsules installed

Duke Power Company

OC1-D
OC2-F
OC3-E
OC3-F

GPU Nuclear Corporation

TMI1-D
TMI2-D

B&W Owners Group

A2
A4
CR3-LG2
TMI2-LG1
TMI2-LG2

ANNUAL REPORT OF CHALLENGES TO PRESSURIZER RELIEF
AND SAFETY VALVES IN ACCORDANCE WITH
TECHNICAL SPECIFICATION 6.9.1.5.(e).

REPORTING PERIOD
JANUARY 1, 1992 TO DECEMBER 31, 1992

There were two (2) challenges to the Pressurizer Relief and Safety Valves in 1992. These 2 challenges were associated with normal plant startup procedures.

ANNUAL REPORT OF
FACILITY CHANGES, TESTS, AND EXPERIMENTS
AT CRYSTAL RIVER UNIT 3
IN ACCORDANCE WITH 10 CFR 50.59(b)

REPORTING PERIOD
JANUARY 1 - DECEMBER 31, 1992

In the attached report, each number refers to the Safety Evaluation questions listed below:

1. Is the probability of occurrence or the consequences of an accident or malfunction evaluated in the FSAR increased?

   YES ___ NO ___

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

   YES ___ NO ___

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

   YES ___ NO ___

10 CFR 50.59 EVALUATIONS PERFORMED
ON PLANT MODIFICATIONS
IN ACCORDANCE WITH MAR PROCEDURES

FACILITY MODIFICATION
MAR 86-05-08-01
SMART AUTOMATIC SIGNAL SELECTION

SAFETY EVALUATION

1. NO

This modification implements automatic signal selection of redundant parameter signals used in the ICS/NNI Systems for control. Redundant sensors/signals have been added where required for automatic selection. Implementation of Automatic Signal Selection will reduce the probability of plant trips or transients resulting from failure of the sensor/signal providing input to the control loop. In addition this modification implements buffering of selected "RY" indicators which could load the control signal it is displaying during a loss of AC power to the indicator. The additional buffering will prevent degradation of the control signal and therefore will reduce the probability of plant trips or transients which could occur due to loss of AC power to the "RY" indicator(s).

2. NO

This modification greatly improves reliability of control signals used by the ICS/NNI for plant operation and will reduce the probability of plant trips and transients which would result from failure of the controlling sensor/signal.

3. NO

The technical specifications do not address requirements for any changes being implemented by this modification.

PLANT MODIFICATION
MAR 86-05-08-02
ICS/NNI REMOVAL OF UNUSED/INOPERABLE HARDWARE

SAFETY EVALUATION

1. NO

This modification consists of five independent changes to the Non-Safety Related ICS/NNI. Each change is addressed separately by the following evaluation.

a. RC Flow & ULD Runback

The RC Flow & ULD Runback Circuits are being removed from the ICS. The RC Flow & ULD Runback Circuits have been previously disconnected and superseded by RC Pump Status Signals already in use in the ICS Unit Load Demand.

Removal of the RC Flow & ULD Runback Circuits and determination of associated cables will prevent a failure within the subject equipment or cables from affecting the ICS Control Signals.

b. Grid Frequency Error Circuits

The Grid Frequency Error Circuits are being removed from the ICS. The Grid Frequency Error Circuits have been previously detuned to the extent that they are ineffective. This detuning has not caused any problems.

Removal of the Grid Frequency Error Circuits and determination of the associated cables will prevent a failure within the subject equipment or cables from affecting the ICS Control Signals.

c. Auto Dispatch Circuits

The Auto Dispatch Function and hardware is being removed from the Integrated Control System. This function was intended for remote dispatch control of the plant. The Auto Dispatch Function has never been made operational for Load Control of the Plant.

Removal of the Unused Auto Dispatch equipment and determination of associated cables will prevent a failure within the subject equipment from affecting the ICS Control Signals.

d. Anticipatory Reactor Trips (ARTS) Non-1E

The Non-1E ARTS located in the ICS and NNI Systems is being removed. The Non-1E ARTS is no longer used since its function has been superseded by the fully qualified 1E ARTS System installed by MAR 79-10-86.

Removal of the Non-1E ARTS equipment and determination of associated cables will prevent a failure within the subject equipment from affecting the ICS/NNI Control Signals.

e. PM Indicators

The PM Indicators in the 4160 ES Switchgear Room at elevation 108' which receive their signals form the NNI System, are being removed. The PM indicators are no longer used and their function has been superseded by the Remote Shutdown System instrumentation installed by MAR 77-07-01. Removal of the PM Indicators and determination of the associated cables will prevent a failure of the subject indicators or cable from affecting NNI Signal used for Control and/or Indication.

2. NO

This modification consists of five independent changes to the Non-Safety Related ICS/NNI. Each change is addressed separately by the following evaluation.

a. RC Flow & ULD Runback

Removal of the RC Flow & ULD Runback equipment and determination of associated cables eliminates the possibility of a failure within the subject equipment or cables from affecting control of the ICS.

The RC Flow & ULD Runback Circuit inputs have been previously disconnected, therefore control of the plant is unaffected by this change.

b. Grid Frequency Error Circuits

Removal of the Grid Frequency Error equipment and determination of associated cables eliminates the possibility of a failure within the subject equipment or cables from affecting control of the ICS.

The Grid Frequency Circuits have been previously detuned to the extent that they have no control over the ICS, therefore control of the plant is unaffected by this change.

c. Auto Dispatch Circuits

Removal of the Auto Dispatch equipment and determination of associated cables eliminates the possibility of a failure within the subject equipment or cables from affecting control of the ICS.

The intended function of the Auto Dispatch Circuits has never been operational, therefore control of the plant is unaffected by this change.

d. Anticipatory Reactor Trip (ARTS) Non-1E

Removal of the Non-1E ARTS equipment and determination of the associated cables eliminates the possibility of a failure within the subject equipment or cables from affecting the ICS/NNI signals used for control. The function performed by the Non-1E ARTS has been superseded by a fully qualified 1E ARTS System installed by MAR 79-10-86. Therefore control of the plant is unaffected by this change.

e. PM Indicators

Removal of the PM indicators in the 4160 ES Switchgear Room at elevation 108' and determination of the associated cables eliminates the possibility of a failure in the indicator or its cable from affecting the NNI System Signal used for control and/or indication.

The information provided by the PM indicators being removed is being provided by the Remote Shutdown System installed by MAR 77-07-01. Therefore information available to plant operating personnel is unaffected by this change.

3. NO

This modification will not reduce the Margin of Safety, as defined in the basis for any technical specification. The Technical Specifications do not address requirements for any of the changes being implemented by this modification.

This modification consists of five independent changes to the Non-Safety Related ICS/NNI. Each change is addressed separately by the following evaluation.

a. RC Flow & ULD Runback

The RC Flow & ULD Runback, Input Signals having been previously disconnected, does not exert any control over the ICS, therefore this function is not required for plant operation.

b. Grid Frequency Error Circuits

The Grid Frequency Error Circuits do not, under current plant operation, exert any control over the ICS, therefore this function is not required for plant operation.

c. Auto Dispatch Circuits

The Auto Dispatch function being removed by this modification has never been operational, therefore no replacement is required.

d. Anticipatory Reactor Trips (ARTS) Non-1E

The Non-1E ARTS functions being removed by this modification have been superseded by a fully qualified 1E ARTS System installed by MAR 79-10-86. The 1E ARTS provides the necessary signals for Plant Shutdown.

e. PM Indicators

The PM Indicators being removed by this modification have been superseded by the Remote Shutdown System installed by MAR 77-07-01. The Remote Shutdown System provides the information required for Plant Operations outside the Control Room.

FACILITY MODIFICATION
MAR 86-05-08-04
NEUTRON FLUX SIGNALS TO SASS

SAFETY EVALUATION

1. NO

This modification consists of a change in the neutron flux signal auctioneering used by the Integrated Control System (ICS) for control. Redundant auctioneered power range neutron flux signals will be provided to the ICS, and in conjunction with automatic signal selection (MAR 86-04-08-01) will eliminate plant trips or transients resulting from the loss of a single neutron flux signal from RPS.

2. NO

This modification does not create the possibility for an accident or malfunction of a different type than any previously evaluated in the FSAR. This modification greatly improves reliability of the neutron flux signal to the ICS and will eliminate plant trips and transients from the loss of a single neutron flux signal from the RPS.

3. NO

This modification will not reduce the margin of safety, as defined in the basis for any technical specification. The technical specifications do not address requirements for any changes being implemented by this modification.

PLANT MODIFICATION
MAR 86-05-08-08A
DELETION OF STARTUP FEEDWATER FLOW CORRECTION FROM ICS

SAFETY EVALUATION

1. NO

The Integrated Control System (ICS), which contains the main feedwater flow control circuits being modified by this MAR, has been previously classified as Non-Safety Related, and thus does not represent equipment important to safety. As demonstrated by the results of MAR T86-05-08-08, the deletion of these circuits will not degrade automatic main feedwater control.

2. NO

The ICS has been previously classified as Non-Safety Related. This modification reduces the probability of a loss of main feedwater event and the subsequent plant trip or transient by reducing the quantity of ICS modules and eliminating the dependence on another flow signal which is not redundant (i.e., Startup Feedwater Flow).

3. NO

The ICS is not relied upon for any safety margin in the Technical Specification.

PLANT MODIFICATION
MAR 87-10-09-01A
ASV-5/ASV-204 POWER SEPARATION

SAFETY EVALUATION

1. NO

ASV-5 and ASV-204 are motor operated valves having identical functions of supplying steam to the turbine driven Emergency Feedwater Pump (EFP-2). Since EFP-2 is the ES "B" channel pump, ASV-5 and 204 were electrically connected in parallel to a common 250/125 VDC ES "B" channel power and control source. This modification electrically separates ASV-204 from ASV-5 and repowers ASV-204 from 250/125 VDC ES "A" channel power. Also, separate control room controls and separate "A" channel EFIC interlocks are being provided for ASV-204. Automatic control logic of ASV-204 has not changed. Therefore, the probability of an occurrence of the consequences of an accident or malfunction of equipment important to safety as previously evaluated in the FSAR is not increased since the logic of automatically opening ASV-204 whenever the EFIC System calls for emergency feedwater has not been altered. The reliability of EFP-2 has actually been increased because with this modification either "A" or "B" train power will control and operate one of the steam inlet valves to EFP-2 as opposed to both valves being "B" train powered.

2. NO

The electrical separation of ASV-204 from ASV-5 does not impact the design function of either valve to supply steam to the EFP-2 turbine. Power and control for ASV-5 is not affected by this modification and ASV-5 retains its automatic control logic, remote manual control, local manual control and remote shutdown isolation and control. ASV-204 is being powered from the redundant power channel, and will be provided with its own remote and local manual control and with separate EFIC interlocks for automatic operation. The type of manual control and automatic operation of ASV-204 is the same as for ASV-5. Therefore, based on the above, the possibility for an accident or malfunction of a different type than any previously evaluated in the FSAR is not created.

3. NO

This modification enables the turbine driven Emergency Feedwater Pump (which is the "B" channel pump) to be operational even if a failure should occur on the "B" channel power system for which shutdown operation would be via the "A" channel systems. With this capability, the turbine driven EFW pump is able to operate and share the EFW requirements with the "A" channel motor driven EFW pump. This will reduce the electrical load on the "A" channel diesel generator for the condition of an ES actuation coincident with a loss-of-offsite-power and failure of the "B" channel power system. Consequently, with this modification the margin of safety, as defined in the basis for any Technical Specification, is not reduced. It is actually enhanced because of the increased availability of the turbine driven Emergency Feedwater Pump.

SAFETY EVALUATION

1. NO

The MAR changes the logic for the cooling water isolation valves, SWV-79, -80, -81, -82, -83, -84, and -86 to close the valves when the Nuclear Services Closed Cycle couling Waster (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 psig) 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 va[illegible] 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, -43, or -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 in the FSAR.

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 the FSAR.

With respect to the SBLOCA, the cooling water to the RC Pumps is provided by the SW System during both normal and emergency plant condition. 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 the FSAR.

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 change the containment isolation functions of the SW System penetrations which serve the RBCUs.

The consequences of the Station Blackout Accident, 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 the FSAR.

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 vales 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. Therefore, this modification does not increase the consequences of accidents previously evaluated in the FSAR.

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. Therefore, the proposed activity does not increase the probability of occurrence or a malfunction of equipment important to safety evaluated in the FSAR.

The equipment associated with this modification is safety-related.

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. Therefore, the consequences of a malfunction of equipment important to safety previously evaluated in the FSAR will not increase.

2. 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 RBCUs, and to meet the criteria for Reactor Building Isolation. In that the SW System lines to the RC Pumps and the RBCUs 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. Therefore, the possibility for an accident of a different type than any previously evaluated in the FSAR has not been created by this modification.

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. Therefore, this modification has not created the possibility for a different type of malfunction of equipment important to safety than any previously evaluated in the FSAR.

3. NO

T.S. Section 3/4.6.2, Containment Cooling Systems, discusses that function of the Containment Cooling System, and instructs that the cooling water flow at a rate of 500 GPM or greater shall be verified to each unit cooler at least once per 31 days.

T.S. Section 3/4.6.3, Containment Isolation Valves, discusses the function of the containment isolation valves, and instructs that they shall be demonstrated to be operable during plant operating modes 5 and 6 or at least once per 18 months.

T.S. Section 3/4.7.4, Nuclear Services Closed Cycle Cooling System, discusses the function of the Nuclear Services Closed Cycle Cooling System, and instructs that each manual, power operated, or automatic valve in this system shall be verified to be in its correct position at least once per 31 days.

Since this modification does not affect any of the above Tech Spec parameters, there is no effect on the Tech Spec Bases.

PLANT MODIFICATION
MAR 90-04-03-01
EMERGENCY FEEDWATER PUMP HOTWELL SUCTION STRAINER

SAFETY EVALUATION

1. NO

The Hotwell Suction Strainer will prevent debris from inside the Hotwell from entering the Emergency Feedwater Pump suction piping. The debris could not only clog the EF Pump intake line, which would result in a loss of the Hotwell as Emergency Feedwater source, but it could travel to the Emergency Feedwater Pumps and could cause a failure of the pumps. This modification will have no effect on the operation or performance of the Emergency Feedwater System or the Condensate System. The pressure drop associated with the partially clogged strainer during full emergency feedwater flow is <0.01 psi. There will be no effect on the emergency feedwater pump NPSH. See FPC Calculation M90-1015 for additional information. Installation of the strainer in the non-safety Hotwell will not effect the performance of the Condensate System as it will be located away from the Condensate Pump intake and will not affect the volume of condensate in the Hotwell. Installation of the strainer will not affect the probability of an occurrence or the consequence of an accident or malfunction of equipment important to safety as previously evaluated in the FSAR.

2. NO

Installation of the Hotwell Strainer will not affect the performance or operation of the Emergency Feedwater System or the Condensate System because the differential pressure across the strainer is <0.01 psi and there will therefore be no effect on the EF pumps' NPSH. Clogging a portion of the strainer will not challenge the strainer's structural integrity because force caused by the pressure drop on the largest side of the strainer is <4 pounds. This force is much less than the ultimate tensile strength of the wire. The Strainer will prevent debris from entering or clogging the EF Pump suction which is located in the Hotwell. This passive piece of equipment is located in a non-safety location and will not create the possibility for an accident or malfunction of a different type that any previously evaluated in the FSAR.

3. NO

Installation of the Emergency Feedwater suction strainer will prevent Hotwell debris from entering the Emergency Feedwater Pump suction piping and the pumps themselves. No impact on the operation or the performance of the Emergency Feedwater or the Condensate System will be created by the strainer installation because the strainer cannot become clogged to a point which could challenge the EF pumps' NPSH or the structural integrity of the strainer. No effect on the Margin of Safety as defined in the basis for any Technical Specification will be experienced.

PLANT MODIFICATION
MAR 90-07-03-01
SBO-NITROGEN BACKUP FOR ADV's

SAFETY EVALUATION

1. NO

This MAR affects the Nitrogen (NG), Instrument Air (IA), and Main Steam (MS) Systems an the Turbine and Intermediate Buildings. As will be shown below, the MAR does not affect any structures, systems or components in a manner that will increase the probability of an accident or a malfunction of equipment important to safety.

The nitrogen bottles, tubing, and valves are considered part of the NG System; however, they are independent of the existing NG System and therefore have no effect on it.

The Nitrogen Backup System is connected to the IA System upstream of the air supply to the actuators of the ADV's, MSV-25, and MSV-26. The Nitrogen Backup System is normally isolated from the ADV's and should not affect their operation. Further, the Nitrogen Backup System provides a source of gas to operate the valves, but it does not affect the control circuit of the ADV's. A relief valve is provided in the Nitrogen Backup System to protect the components of the ADV actuators in the event of a failure of the pressure regulators. Because of these features, this MAR does not increase the probability of a failure of the ADV's. This MAR does not impact the MS System in any other way, and therefore the probabilities of or the consequences of any accident, notably a Main Steam Line Failure and a Steam Generator Tube Rupture, are not increased. This equipment will not prevent the ADV's from opening if required. Check valves are provided to prevent instrument air from being diverted away from the ADV's. Even if the check valves fail, NGV-312 is normally closed so air flow still will not be diverted from the ADV's.

The core drill in the "G" wall between the Turbine and Intermediate Buildings will be done to safety-related standards, sealed with Masterflow non-shrink grout, and will not cut any rebar in the G wall. Therefore the probability of a failure of this safety-related wall is not increased. The drilling of holes in the G wall and other Intermediate Building walls for supports will also be done to safety-related standards. These holes also will not cut any rebar. Therefore, the probability of failure of these safety-related walls is not increased.

2. NO

This MAP noes not create the possibility of a new accident or malfunction of safety-related equipment. The Nitrogen Backup System is normally not in service, so any system failures will not cause an accident. In the event that the system is inadvertently aligned for service, it will not cause a plant accident. The actuators of the ADV's are protected from being over pressurized by the high pressure nitrogen tanks by the pressure regulators, NGV-299 and NGV-308, and the relief valves, NGV-314, NGV-319, and NGV-320. This additional gas supply does not affect the ADV's control system, and

will not cause an inadvertent opening of the ADV's. The high pressure nitrogen bottles, and the high pressure portions of the system upstream of the relief valves, are in the Turbine Building, where there is no safety-related equipment. Any failures of the Nitrogen System will not affect any safety-related equipment in an adverse way.

A failure of the Nitrogen System in the IB also will not cause any new malfunctions of safety-related equipment. The tubing in the IB is designed to "anti-fall down" criteria to prevent any seismically induced systems interactions. Nitrogen is inert, so any release of nitrogen into the IB will not create any environmental problems for safety-related equipment.

3. NO

All of the Technical Specifications were reviewed and none of the equipment affected by this modification, NG System, IA System, or the ADV's are included within the Technical Specifications. Therefore, the margin of safety as defined in the basis for a Technical Specification is not reduced.

PLANT MODIFICATION
MAR 90-08-10-01
SOURCE TUBE CAVITY DOSIMETRY

SAFETY EVALUATION

1. NO

Source Tube Cavity Dosimetry (STCD) hardware is a stand-alone, self-contained system having no fluid nor electrical system interfaces. The STCD hardware is not part of any safety system required for accident mitigation for events identified in the Final Safety Analysis Report. The amount of Aluminum added into containment falls within the boundaries of calculation M91-0022.

2. NO

STCD hardware is a stand-alone, self-contained system have no fluid nor electrical system interfaces. The dosimetry hardware is supported from the bottom of the NI box. The possibility of an accident of a different type than any previously evaluated in the FSAR will not be created.

3. NO

STCD hardware will be used to monitor the RV for neutron fluence. It will not be installed at a pressure boundary and will not be connected to automatic or manually activated control devices. Therefore, it will have no effect on the margin of safety of other existing equipment during plant operations.

PLANT MODIFICATION
MAR 90-08-15-01
MK-B7 FUEL ASSEMBLY

SAFETY EVALUATION

1. NO

The recaged MK-B7 fuel assembly is functionally equivalent to the present MK-B4Z fuel assembly in Crystal River's core. The interface with the rest of the core is identical. The small physical differences of the recaged MK-B7 compared to the MK-B4Z do not increase the probability of occurrence, or the consequences of any FSAR accidents.

2. NO

The recaged MK-B7 fuel assembly is functionally equivalent to he present MK-B4Z fuel assembly in Crystal River's core. The interface with the control rods are identical. The small physical differences between the recaged MK-B7 and the MK-B4Z do not result in any accidents that differ from the FSAR accidents.

3. NO

The recaged MK-B7 fuel assembly is functionally equivalent to the present MK-B4Z fuel assembly. The DNBR margin is not altered, so the margins to setpoints are not altered. The small physical differences between the two fuel assemblies do not result in a reduction of any Technical specification Margin of Safety.

PLANT MODIFICATION
MAR 91-03-04-01
CONDENSATE STORAGE TANK LEVEL

SAFETY EVALUATION

1. NO

This modification will remove CD-67-LT string from service. This instrument was for level indication of CDT-1 and was redundant to CD-67-LT1 string.

Until EFT-2 was installed, CDT-1 was the main Emergency Feedwater source and redundant instrumentation was required. CDT-1 is now a backup source of Emergency Feedwater and no longer requires redundant indication.

The removal of this obsolete, unreliable, Non-Safety Related instrument will not increase the probability of the occurrence or the consequences of an accident or malfunction of equipment important to safety as evaluated in the FSAR.

2. NO

The deletion of CD-67-LT string does not create the possibility of an accident or malfunction different from those evaluated in the FSAR because the CDT-1 level indication will still be available from CD-67-LT1. CDT-1 is not required to have redundant indication.

3. NO

The instrumentation for CDT-1 is not a part of CR-3 Technical Specifications. The instrument being removed is redundant to CD-67-LT1. There is no longer any requirement for redundant indication on CDT-1 as it has been replaced as the primary Emergency Feedwater Source by EFT-1. If CDT-1 were to be made the <u>temporary</u> source for Emergency Feedwater because of temporary problems with EFT-2, CD-67-LT1 would satisfy any surveillance requirement imposed without a reduction in the margin of safety.

10 CFR 50.59 EVALUATIONS PERFORMED
ON PLANT PROCEDURES
IN ACCORDANCE WITH ADMINISTRATIVE INSTRUCTIONS

PLANT PROCEDURE

AI-400B

SAFETY EVALUATION

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 pick-up during block 1. Calculation E-88-0001 and E-91-0027 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 85% voltage is conservative.

2. NO

This revision does not change the testing method or equipment line ups as it presently exists. Instead, it ensures that the system response times are within the time limits assumed in the Safety Analysis.

3. NO

This revision provides assurance that the ESFAS action required for each channel is completed within the time limit assumed in the Safety Analysis.