

**Florida
Power**
CORPORATION

W. P. STEWART, DIRECTOR
POWER PRODUCTION

January 10, 1979

Mr. Robert W. Reid
Chief, Operating Reactors Branch #4
Division of Operating Reactors
U.S. Nuclear Regulatory Commission
Washington, DC 20555

SUBJECT: Florida Power Corporation
Docket No. 50-302
Operating License No. DPR-72

Dear Mr. Reid:

In your letter of November 29, 1978, concerning containment purging during normal plant operation, you identified two events which occurred during power operation at Millstone Unit No. 2 and Salem Unit No. 1. Specifically, your letter addressed the fact that containment purging had occurred at these facilities with the safety actuation isolation signals to the purge isolation valves manually overridden and inoperable.

As a result of these events, you requested Florida Power Corporation to review the override circuitry used at Crystal River Unit 3 and to provide justification for continuing unlimited purging at Crystal River Unit 3.

In response to your request, Florida Power Corporation hereby submits for your staff's review forty (40) copies of the results of our review of the override circuitry at CR#3 as well as our justification, based on this review, for continuous purging at CR#3. Our response is contained in Enclosure 1 to this letter.

Should you or a member of your staff require any further discussion of this matter, please contact this office.

Very truly yours,

FLORIDA POWER CORPORATION

W.P. Stewart

File: 3-0-3-a-3

Enclosure
WPS/ECS/hlcT02
(1/9D3)

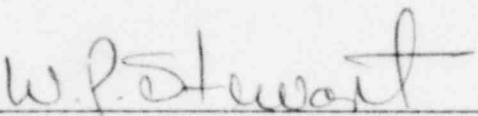
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STATE OF FLORIDA

COUNTY OF PINELLAS

W.P. Stewart states that he is the Director, Power Production, of Florida Power Corporation; that he is authorized on the part of said company to sign and file with the Nuclear Regulatory Commission the information attached hereto; and that all such statements made and matters set forth therein are true and correct to the best of his knowledge, information and belief.



W.P. Stewart

Subscribed and sworn to before me, a Notary Public in and for the State and County above named, this 10th day of January, 1979.



Notary Public

Notary Public, State of Florida at Large,
My Commission Expires: August 24, 1979

RESPONSE TO THE NRC REQUEST FOR ADDITIONAL INFORMATION

I. R.B. Purge System

We have reviewed the above referenced NRC letter and offer the following response. Our Reactor Building Purge supply and exhaust valves are arranged in pairs such that the supply and exhaust purge systems each contain a solenoid operated pneumatic valve outside of the Reactor Building (AHV-1A and AHV-1D) and a corresponding motor operated valve in series inside of the Reactor Building (AHV-1B and AHV-1C, respectively). All four valves have three methods of closure; namely, an engineered safety feature signal, a high radiation signal or a manual closed signal.

The Reactor Building Isolation signal is a similar design to the rest of the engineered safety feature system utilizing a two out of three actuation signal from the pressure switches used to sense Reactor Building pressure. Whenever the Reactor Building pressure exceeds 4 psig, the pressure switches actuate causing an actuation matrix to become energized and a closed signal interrupts the control power to the solenoid resulting in spring closures on the pneumatic valves and a closed signal completes the control power circuit to energize the closure contactor for the motor operated valves. An auxiliary relay from the closure circuit on the motor operated valves also blocks the opening circuit of those valves. The CR#3 design also prevents bypassing of an engineered safety feature signal until after an actuation of the ESF system has occurred. In addition, there are no individual bypass circuits to operate any of these valves once an engineered safety feature signal has closed the valves.

The second signal causing reactor building purge isolation is a high radiation signal from the Reactor Building purge duct radiation monitor (RM-A1). Upon receipt of high radiation signal, an auxiliary relay is energized to interrupt the control power to the solenoids on the pneumatic valves resulting in spring closures and the high radiation signal auxiliary relay also energizes motor operated valve closure circuits and blocks the motor operator valve open circuitry. The high radiation signal of a single monitor can only be bypassed at the radiation monitoring panel. This bypass then actuates an alarm on the control board and lights an amber light on the radiation monitoring panel. This bypass does not directly eliminate the other ECCS actuated functions related to containment building purge isolation.

Florida Power Corporation is presently modifying this bypass switch from a push/lock switch to a spring return type, thereby requiring continual operator action to maintain the override. Additionally, the monitor purge switch will also be converted to a spring return (to normal operation) to prevent disabling of the monitor by isolating the gas channel from the process flow by inadvertently leaving the monitor purge switch in the purge position.

These Reactor Building isolation valves can also be closed manually, either from the Control Board engineered safety featured panels or locally near the solenoid valves or at the motor control center for the motor operated valves.

There is an additional radiation monitor (RM-A6) which alarms when the Reactor Building air sample line radiation value exceeds a high setpoint. This monitor provides a second source of alarm of Reactor Building high radiation conditions.

The above discussion supports FPC's contention that the events that occurred at Millstone Unit No. 2 and Salem Unit No. 1 are precluded by the CR#3 design, in that the engineered safety feature signal supplied to the R.B. Purge valves cannot be bypassed while purging the building and that bypassing the high radiation signal does not prevent R.C. purge isolation during postulated accident conditions.

We also conducted a review of the design bases of the R.B. purge isolation valves and have concluded that they will close against the dynamics forces of a design basis LOCA. The design requirements, as described in Section 5.3.3.1 of the FSAR, which were applied to the four R.B. purge isolation valves are:

1. Exterior valves:
 - a. Must be capable of closing against a differential pressure of 55 psig
 - b. Must close fully in 2 seconds
 - c. When closed the valve must seal bubbletight, that is, no air bubbles appear in a pool of water, with 63.3 psig air pressure applied across the closed face.
2. Interior valves
 - a. Must be capable of closing against a differential pressure of 55 psig
 - b. Must close fully in 5 seconds
 - c. When closed, the valve must seal bubbletight under the same conditions as item 3 above.

The highest peak Reactor Building pressure calculated for the design basis LOCA at CR#3 is 49.6 psig (FSAR Section 13.2.2.5.6) which is within the design requirements for the R.B. purge isolation valves.

In addition, restriction of purging operation at CR#3 will render the RCS leakage detection capabilities of RM-A6 required in Technical Specification 3/4 4.6.1 a and c inoperable because the increased radioactivity in the Containment Building will saturate the detector. In accordance with this specification, if RM-A6 is inoperable, CR#3 must be in at least Hot Standby within 6 hours and in Cold Shutdown within the next 30 hours.

Elimination of continuous purging or restricting the purging to 90 hrs/yr. will result in increased radioactivity in the Containment Building which will increase the dose to personnel working at the plant. This is in contradiction of Florida Power Corporation's ALARA position. It is also the opinion of Florida Power Corporation that due to the buildup of radioactive material in the Containment Building during periods of little or no purging, there is some question as to the soundness of dumping in a batch versus the discharge of small quantities of radioactive materials over a year's time. Even with the advantage of decay with holdup, it is uncertain that the dose to the public could be reduced with no or limited purging during operation, requiring purging at higher activity levels during Modes 5 and 6, as compared to continuous low level discharge during all modes of operation.

It is the position of Florida Power Corporation that the above discussion provides adequate justification that the events identified in Mr. Reid's letter of November 19, 1978 cannot occur at CR#3 and that continuous purging at CR#3 is in the best interest of the safety and welfare of the public and employees of Florida Power Corporation.

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II. Review of the Safety Actuation Signal Circuits

We have also reviewed the design of the safety actuation signal circuits at CR#3, to ensure that the bypassing of one safety actuation signal does not also cause the bypass of any other safety actuation signal.

The safety actuation circuits at CR#3 are comprised of the Engineered Safety Feature Actuation and Reactor Protection Systems.

As stated in Part I of this response, the CR#3 design prevents bypassing of an engineered safety feature signal until after an actuation of the ESFA system has occurred and the operator wishes to take manual control of certain safety equipment.

The Reactor Protection System utilizes a key switch in each protection channel for changing the system logic from a two out of the four total channels to a two out of three coincidence. Operation of the key switch:

1. Blocks the trip action of the associated protection channel;
2. Renders the remaining key switches ineffective in blocking trip action of their respective protection channel.

Thus, one and only one protection channel may be bypassed at a time.

With one reactor Protection System Channel bypassed startup and power operation may proceed provided the following technical specification requirements are satisfied:

1. Within 1 hour:
 - a. Place the inoperable channel in the tripped condition, or
 - b. Remove power supplied to the control rod trip device associated with the inoperative channel placing it in the tripped condition.
2. One additional channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.1.1, and the inoperable channel above may be bypassed for up to 30 minutes in any 24-hour period when necessary to test the trip breaker associated with the logic of the channel being tested per Specification 4.3.1.1. The inoperable channel above may not be bypassed to test the logic of a channel of the trip system associated with the inoperable channel.

Also, whenever a Reactor Protection System Channel is placed in the bypass mode, this condition is annunciated in the Control Room.

In light of the above discussion, we have concluded that the operation of the bypass circuitry at CR#3 will affect no safety function other than those previously analyzed.