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NUCLEAR PRODUCTION DEPARTMENT

September 2, 1981

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U.S. NUCLEAR REGULATORY COMMISSION

U.S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation
Washington, D. C. 20555

Attention: Mr. Harold R. Denton, Director

Dear Mr. Denton:

SUBJECT: Grand Gulf Nuclear Station
Units 1 and 2
Docket Nos. 50-416 and 50-417
File 0260/L-334.0
Transmittal of Proposed FSAR Changes and Responses to NRC Questions
Reference: NRC Question 031.67
AECM-81/341

Reference:

1. Instrumentation and Control Systems Branch Question 031.67.
2. Containment Systems Branch, Informal questions in regards to the In-Plant Safety/Relief Valve Test Program.
3. Auxiliary Systems Branch, Question 10.28.
4. Instrumentation & Control Systems Branch, Informal request for information pertaining to Reactor level reference leg boil-off.
5. Reactor Systems Branch, Informal request for information pertaining to LOCA-Recirc. Valve Failure.

In response to your request for additional information, Mississippi Power & Light Company is submitting the enclosed material updating information pertaining to references above.

This information, unless otherwise noted, represents changes to the Grand Gulf Nuclear Station Final Safety Analysis Report (FSAR).

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These proposed FSAR changes will be incorporated into a forthcoming amendment to the FSAR. If you have any questions or require further information, please contact this office.

Yours truly,

L. F. Dale
Manager of Nuclear Services

RFP/JGC/JDR:cm

Attachment:

1. Instrumentation and Control System Branch - Question & Response 031.67

2. Containment Systems Branch - Informal Questions pertaining to the In-Plant Safety/Relief Valve Test Program

3. Auxiliary Systems Branch - Question and Response 10.28

4. Instrumentation & Control Systems Branch - Informal request for information pertaining to Reactor level, reference leg boil-off.

5. Reactor Systems Branch - Information request for information pertaining to LOCA-Recirc. Valve Failure

cc: Mr. N. L. Stampley
Mr. R. B. McGehee
Mr. T. B. Conner
Mr. G. B. Taylor

Mr. Victor Stello, Jr., Director
Office of Inspection & Enforcement
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

031.67 (7.3) The accident analysis in Chapter 15 takes credit for the pressure relief available through the automatic sequencing and operation of the safety relief valves. Justify the exclusion of the instrumentation and controls for the relief valves from Chapter 7 in general and Section 7.3 in particular.

RESPONSE

The response to this question is provided in Section 7.3 which has been revised to include Subsection 7.3.1.1.1.4.12, "Safety Relief Valves". Controls and Instrumentation are explained therein.

Additional information is also provided in the response to Question 021.6 on Page Q & R 6.2-9, Q & R 6.2-9a, Figures 021.6-1 and 2, and Table 5.2-2.

7.3.1.1.1.4.12 Safety Relief Valves

7.3.1.1.1.4.12.1 System Identification

The nuclear pressure relief system is designed to prevent overpressurization of the nuclear system that could lead to the failure of the reactor coolant pressure boundary.

7.3.1.1.1.4.12.2 SRV Equipment Design

The automatic safety-relief system consists of redundant reactor pressure instrument channels arranged in separate logics that control separate solenoid-operated air pilots on each valve. These pilot valves control the pneumatic pressure applied to an air cylinder operator. An accumulator, one on each valve, is included with the control equipment to store the pneumatic energy for relief valve operation. SRV's are initiated by reactor vessel pressure. The SRV's are divided into 3 pressure groups. The first group consists of 1 valve which opens when vessel pressure exceeds 1103 psig. The second group consists of 10 valves which open when vessel pressure exceeds 1113 psig. The third group consists of 9 valves which open when vessel pressure exceeds 1123 psig. Control cables from the vessel pressure sensors lead to two separate logic cabinets where the redundant logics are formed. Separate station batteries power the electrical control circuitry. The power supplies for the redundant logics are separated to limit the effects of electrical failures. Power sources for the SRV solenoids and logics are the same as the ADS Divisions 1 and 2.

7.3.1.1.1.4.12.3 Initiating Circuits

Reactor pressure is detected by four pressure transducers (two for each division), which are located in the containment. The SRV control logic requires a two-out-of-two trip (per division) on vessel pressure to open the SRV's. This also prevents inadvertent SRV actuation (of more than one valve) due to single failure. The SRV control logic is arranged such that no single failure will prevent SRV actuation or cause more than one SRV to inadvertently actuate.

7.3.1.1.1.4.12.4 Logic and Sequencing

Two-out-of-two reactor vessel pressure instrument signals/trips are required to initiate the safety-relief valves. High vessel pressure indicates the need for SRV actuation to prevent nuclear steam overpressure.

After receipt of the initiation signal, each of the two solenoid pilot air valves on each safety relief valve is energized. Either or both solenoid actuations allow pneumatic pressure from the accumulator to be applied to the air cylinder operator. The air cylinder operator holds the valve open. Lights in the control room on the Reactor Core Cooling Benchboard indicate when the solenoid-operated pilot valves are energized to open a safety relief valve. The SRV's remain open until system pressure drops below the reclose setpoint.

Manual opening of each SRV is accomplished by either a control switch in the Division 1 (A logic) portion of the main control room or by a control switch in the Division 2 (B logic) portion of the control room panel.

Two redundant SRV trip systems are provided - Logic A and Logic B. Division 1 sensors for high reactor pressure initiate Logic A, and Division 2 sensors initiate Logic B. The components are mounted in two different cabinets.

The SRV Logic A actuates the "A" solenoid pilot valve on each SRV. The SRV Logic B actuates the "B" solenoid pilot valve on each SRV. Actuation of either solenoid-pilot valve causes the SRV to open to provide depressurization.

7.3.1.1.1.4.12.5 Bypasses and Interlocks

Bypasses and interlocks are not utilized in the SRV function.

7.3.1.1.1.4.12.6 Redundancy and Diversity

The SRV logic is initiated by high reactor pressure. The initiating circuits for each of these variables is redundant, as explained in the circuit description of this section. Diversity is not provided.

7.3.1.1.1.4.12.7 Actuated Devices

All relief valves are actuated by three methods:

a. Automatic action resulting from the logic chains in either Division 1 or Division 2 trip system actuating;

b. Manually by the operator;

c. Mechanical actuation as a result of high reactor pressure.

7.3.1.1.1.4.12.8 Separation

SRV logic is a Division 1 (Logic A) and Division 2 (Logic B) system. Each relief valve can be actuated by either of two solenoid pilot valves supplying air to the relief valve air piston operators. One of the solenoid pilot valves is operated by Logic A and the other by Logic B. Logic circuitry, manual controls, and instrumentation are designed so that Division 1 and Division 2 electrical separation is maintained.

7.3.1.1.1.4.12.9 Low-Low Setpoint Logic

In order to assure that no more than one relief valve reopens following a reactor isolation event, 6 safety relief valves are provided with lower opening and/or closing setpoints. The setpoints for the low-low setpoint logic are listed in the Technical Specifications and override the normal setpoints following the initial opening of more than one relief valve and act to hold these valves open longer, thus preventing more than one single valve from reopening subsequently. This system logic is referred to as the low-low set relief logic and functions such that the containment design basis of one safety relief valve operating

on subsequent actuations is met. For additional information on the low-low set relief functions, please see the response to Question 021.6. This logic is armed from the existing pressure sensors of the second (i.e., @ 1113 psig) normal relief setpoint group. When these sensors trip, low-low-set logic automatically seals in the control circuitry of the selected valves and actuates a control room annunciator. The low-low setpoint logic remains sealed in until manually reset by the operator.

Once sealed in, the low-low set logic acts to hold selected relief valves open past their normal reclosure point until the pressure decreases to a predetermined "low-low" reclosing setpoint. Thus these selected valves remain open longer than the other safety relief valves. This extended relief capacity consumes sufficient energy such that no more than one valve should reopen a second time. Also, the seal-in logic provides the first two low-low set valves with new reopening setpoints which are lower than their original setpoints to remove decay heat. These two valves provide redundancy in case of single valve failure. Transient analysis reveals that only the first valve will reopen on secondary transient peaks unless it failed. In that event, the second valve will act as back-up. (See Table 5.2-2) and Q/R Section 021.6, Page Q&R 6.2-9, Amendment 47.)

The low-low set logic is designed with the same redundancy and single failure criteria as the safety relief valve logic; i.e., no single electrical failure will: (1) cause inadvertent seal-in of low-low set logic, (2) prevent any low-low set valve from opening, or (3) cause more than one valve to inadvertently open or stick open.

The valves associated with low-low set are arranged in three secondary setpoint groups or ranges (low, medium, high). The "low" and "medium" pressure ranges consist of one valve each, having altered reopen and reclose setpoints independently adjustable. These two valves are set considerably lower than their normal SRV setpoints. The remaining valves are simultaneously controlled by the "high" range sensors which have an independently adjustable reclose setpoint. The normal SRV opening setpoint is retained for this valve group though reclose is extended in the low-low set operating mode.

The sensors are arranged in two trains for each division. These conform to safety relief logics "A" and "E" for Division 1 and "B" and "F" for Division 2. Thus, the single-failure criterion is maintained because two-out-of-two logic trains (per division) are required to open the valves and one-out-of-two in each division acts to reclose them. The low range sensors which control the first valve are placed in logic E (F) and the medium range sensors which control the second valve are placed in logic A (B). The highest pressure sensors which are used for arming and sealing in low-low set logic, control five valves simultaneously. Therefore, these are also arranged in redundant two-out-of-two (A·E) + (B·F) logic to maintain single failure proof integrity.

7.3.1.1.1.4.12.10 Testability

The SRV system has two complete logics, one in Division 1 and one in Division 2. Either one can initiate depressurization. Each logic has

two trains, both of which must operate to actuate the SRV. The SRV instrument channels signals are tested by cross comparison between the channels which bear a known relationship with each other. Meters indicating vessel pressure for each instrument channel are mounted in the logic cabinets. The instrument channel setpoints may be verified by introducing a test signal to move the signal towards trip. The setpoint is verified by observing the meter and the indicator light on the output of the instrument channel trip device. Testing does not interfere with automatic operation if required by an initiation signal.

7._ 1.1.1.4.12.11 Environmental Considerations

The solenoid valves and their cables and the safety relief valve operators are the only control and instrumentation equipment for the SRV system located inside the drywell. Equipment located outside the drywell will also operate in their normal and accident environments.

7.3.1.1.1.4.12.12 Operational Considerations

7.3.1.1.1.4.12.13 General Information

The instrumentation and controls of the S system are required for normal plant operations to prevent nuclear system overpressure. When pressure relief action is required, it will be initiated automatically by the circuits described in this section.

7.3.1.1.1.4.12.14 Operator Information

A temperature element is installed on the safety relief valve discharge piping several feet from the valve body. The temperature element is connected to a multipoint recorder in the control room to provide a means of detecting safety relief valve leakage during plant operation. When the temperature in any safety relief valve discharge piping exceeds a preset value, an alarm is sounded in the control room. The alarm setting is far enough above normal (rated power) drywell ambient temperatures to avoid spurious alarms, yet low enough to give early indication of significant safety relief valve leakage.

NOTE

Additional editorial changes [illegible] made to Section 7.3 to reflect the addition of subsection [illegible] .1.4.12. These changes, although not provided herein, will [illegible] in a forthcoming FSAR amendment.

Comment 1.(a)* Provide detailed justification or modify the test plan to rectify the absence of water level probe instrumentation.

RESPONSE

Pressure sensor P25 on SRVDL V-12 will provide details of the water level during the relief valve discharge. It should be noted that the next revision to the test plan will include a second low range pressure sensor in the V-12 quencher SRV line. These sensors will record the discharge line pressure subsequent to valve closure. This data will provide the water leg height prior to consecutive valve actuations in the V-12 line. Caorso test data has shown that the water level recovery following the closure of the SRV is not significantly higher than the normal water level for lines with two 10 inch vacuum breakers, in parallel, as installed at Grand Gulf.

*This concern pertaining to the Grand Gulf In-Plant Safety/Relief Valve Test Program was directed to MP&L informally by Containment Systems Branch.

The changes discussed herein will be incorporated into the next revision of the test plan.

This response is provided for information only and will not be incorporated into the FSAR.

Comment 1.(b)* Provide detailed justification or modify the test plan to rectify the absence of any accelerometer instrumentation and any stated intent to examine the structural and equipment response to the SRV loads.

RESPONSE

Fifty-six (56) channels of accelerometers are planned to measure building, piping and equipment response to provide backup information to the measured SRV pressure loads. The accelerometers are distributed to fourteen (14) biaxial locations on the containment structure and two (2) biaxial on the auxiliary building structure. In addition there are three (3) sets of triaxial accelerometers on containment equipment and five (5) sets of triaxial accelerometers at piping locations. As stated on page 5-6 of the Test Plan, evaluations of the accelerometer data will be performed only if the pressure acceptance criteria are exceeded for a particular test.

*This concern pertaining to the Grand Gulf In-Plant Safety/Relief Valve Test Program was directed to MP&L informally by Containment Systems Branch.

This response is provided for information only and will not be incorporated into the FSAR.

Comment 1.(c)* Provide detailed justification or modify the test plan to rectify the absence of any means for determining phasing between bubble entry times during multiple valve actuations.

RESPONSE

A reference "zero" timing pulse will be provided to the data acquisition system via SRV actuation signals from the control room. Referring individual bubble pressure time histories back to this "zero" time reference will provide multiple valve actuation phasing information for the pool pressure sensors. As discussed in response to question 2a certain practical limitations prevent simultaneous bubble entry. The 3 valve test compares the contribution of adjacent valves to predicted contributions at representative phasing conditions. The data is not intended to further evaluate phasing considerations. It should further be noted that NUREG-0763 states that the issue of load superposition and the response of structures and equipment to multiple valve actuations must be resolved generically.

*This concern pertaining to the Grand Gulf In-Plant Safety/Relief Valve Test Program was directed to MP&L informally by Containment Systems Branch.

This response is provided for information only and will not be incorporated into the FSAR.

Comment 1.(d)* Provide detailed justification or modify the test plan to rectify why the test matrix does not address the effect of vacuum breaker performance on SRV loads.

RESPONSE

Specific consecutive valve actuations which require manipulations of the vacuum breakers to retard their performance are not included in the test matrix. As noted in item 1a, redundant vacuum breakers are provided in each line to ensure that a malfunction will not affect the predicted reflood height following an SRV closure. In addition, redundant pressure sensors are installed on SRVDL V-12 to provide indication of water level.

*This concern pertaining to the Grand Gulf In-Plant Safety/Relief Valve Test Program was directed to MP&L informally by Containment Systems Branch.

This response is provided for information only and will not be incorporated into the FSAR.

Comment 1.(e)* Provide detailed justification or modify the test plan to rectify the absence of any instrumentation to monitor for a leaky valve and any indication of how the test matrix would be modified to examine the effect thereof on SRV loads.

RESPONSE

Plant SRV tail pipe temperature monitors and pressure switches provide indications of a "leaky" valve. The test matrix does not include a "leaky" valve test. As stated in NUREG-0763 plant specific tests will not generally include leaky valve actuations. GESSAR 3B demonstrates that loads from leaky valve actuations are bound by other design loads.

*This concern pertaining to the Grand Gulf In-Plant Safety/Relief Valve Test Program was directed to MP&L informally by Containment Systems Branch.

This response is provided for information only and will not be incorporated into the FSAR.

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Comment 1.(f)* Provide detailed justification or modify the test plan to rectify the absence of strain gage instrumentation on quencher arms.

RESPONSE

Attachment of strain gages on the quencher arms was considered unnecessary since previous SRV tests at Caorso performed with X-quenchers similar to Grand Gulf's recorded low dynamic stresses on the quencher arms. In addition, because the quencher support structures are more plant unique in comparison to the various plant X-quencher designs, it is more important to locate strain gages on these structures at points of expected maximum stress.

*This concern pertaining to the Grand Gulf In-Plant Safety/Relief Valve Test Program was directed to MP&L informally by Containment Systems Branch.

This response is provided for information only and will not be incorporated into the FSAR.

Question 2.(a)* How is simultaneous firing of multiple valves to be accomplished? Has any consideration been given to accounting for differences in line length volume to insure simultaneous bubble entry?

RESPONSE

Simultaneous firing of multiple valves in the strict sense is not possible. Our multiple valve tests will be accomplished by reactor operators manually actuating the required SRV's at the same time. The valve opening stroke time tolerances and variations of volumes and distribution of line losses make it essentially impossible to provide individual SRV actuation times for multiple valve tests to insure simultaneous bubble entry into the suppression pool.

*This concern pertaining to the Grand Gulf In-Plant Safety/Relief Valve Test Program was directed to MP&L informally by Containment Systems Branch.

This response is provided for information only and will not be incorporated into the FSAR.

Question 2.(b)* What are the volumes of the test discharge lines? What are the smallest and largest volumes of all lines in the plant?

RESPONSE

The requested line volumes are as follows:

V-3	55.00 cu. ft.	(minimum volume)
V-10	55.15 cu. ft.	(test discharge line)
V-11	57.51 cu. ft.	(test discharge line)
V-12	56.84 cu. ft.	(test discharge line)
V-16	57.78 cu. ft.	(maximum volume)

*This concern pertaining to the Grand Gulf In-Plant Safety/Relief Valve Test Program was directed to MP&L informally by Containment Systems Branch.

This response is provided for information only and will not be incorporated into the FSAR.

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Question 2.(c)* At what power of the reactor are the tests to be performed? Is a systematic variation planned? This would be required to fulfill the stated objective to gather sufficient data so as to "extend ... to design conditions" (p.1-1).

RESPONSE

A systematic variation of the basic test parameter (reactor pressure) is not planned. All matrix tests will be performed at a reactor pressure of 1000 ± 15 psia with power level varying between 50% to 75%. Sufficient tests are included in the test matrix to provide a firm statistical data base which can be compared to analytical predictions at the same plant conditions. Adjustment factors derived by comparing test data to analytical predictions can then be extended to analytical predictions based on design parameters.

*This concern pertaining to the Grand Gulf In-Plant Safety/Relief Valve Test Program was directed to MP&L informally by Containment Systems Branch.

This response is provided for information only and will not be incorporated into the FSAR.

Question 2.(d)* Is the accuracy quoted for strain gages ± 3% of full-scale or of readout? If the former, the accuracy would be unacceptable.

RESPONSE

The strain gage accuracy is ± 3% of the applied strain at the input to the signal conditioning.

*This concern pertaining to the Grand Gulf In-Plant Safety/Relief Valve Test Program was directed to MP&L informally by Containment Systems Branch.

This response is provided for information only and will not be incoporated into the FSAR.

Question 2.(e)* Provide clarification regarding the sense in which the choice of valve open and closed times for CVA's will "bound the conditions expected ... during operating transients" (p. 5-2).

RESPONSE

Based on General Electric's transient analyses of the Grand Gulf reactor, the shortest predicted time interval between SVA and CVA is 50 seconds. The CVA's in the test matrix will, therefore occur at 45 seconds after SVA closure in order to bound this design condition. The test plan will be revised to reflect this time interval. Rev. 0 of the test plan specifies a 60 second time interval which was based on the best information available in January 1980.

*This concern pertaining to the Grand Gulf In-Plant Safety/Relief Valve Test Program was directed to MP&L informally by Containment Systems Branch.

The changes discussed herein will be incorporated into the next revision of the test plan.

This response is provided for information only and will not be incorporated into the FSAR.

Comment 3(a)* It might be useful to add some pressure sensors on the RCIC exhaust and RHR line as back-up for strain measurements. The drag loads determined from the latter will be highly structure dependent and will thus have limited applicability in terms of confirming submerged structure load methodology.

RESPONSE

The SRV in-plant tests to be run at Grand Gulf are to confirm the adequacy of the design of the structure, piping and equipment to the predicted SRV pool dynamic loads. It is not intended that these tests should be used to improve the submerged structure load methodology. The final result of the predicted submerged structure loads is a calculation of anticipated pipe strains and these are being measured directly by the installed strain gages. To adequately demonstrate the differential pressure on the submerged structures would take a considerable number of pressu sensors and would not be beneficial to the principal aims of the te Furthermore, other test data such as Caorso and Monticello indica that the submerged structure load methodology is very conservative and this will be confirmed at Grand Gulf by a comparison of measured to predicted strains.

*This concern pertaining to e Grand Gulf In-Plant Safety/Relief Valve Test Program was directed MP&L informally by Containment Systems Branch.

This response is provided for information only and will not be incorporated into the FSAR.

Comment 3.(b)* It might be useful to add pressure sensors at an azimuth 180° away from at least one active quencher. We do not really believe that there is such a thing as a "line of sight cutoff." If non-zero pressures are recorded at azimuth 304° they will have to be imposed uniformly around the containment unless further attenuation can be demonstrated by direct measurement.

RESPONSE

To show the extent of this pressure field pressure sensors P18 and P20 will be relocated to show the pressure field up to 180° away from the azimuth of quencher V-12. The revised locations for these sensors will be P19 to azimuth 312° (51'6" radius); P18 to azimuth 256° (51'6" radius); and P20 to azimuth 200° (51'6" radius).

*This concern pertaining to the Grand Gulf In-Plant Safety/Relief Valve Test Program was directed to MP&L informally by Containment Systems Branch.

The changes discussed herein will be incorporated into the next revision of the test plan.

This response is provided for information only and will not be incorporated into the FSAR.

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Comment 3.(c)* Someone ought to look into the issue raised by the presence of the RCIC turbine exhaust line. This is a generic concern which applies, it seems to us, to all BWR containments. Although it is not as direct as SRV's and downcomers, this line provides a path for steam from the reactor to enter the suppression pool. The pool dynamic loads thus introduced should be as much of a concern as any of the others that have thus far been addressed.

RESPONSE

Since the steam mass flow rate coming from the RCIC exhaust pipe is only about 3.4% of the SRV steam mass flow rate, the dynamic loads associated with the RCIC exhaust are expected to be small.

*This concern pertaining to the Grand Gulf In-Plant Safety/Relief Valve Test Program was directed to MP&L informally by Containment Systems Branch.

This response is provided for information only and will not be incorporated into the FSAR.

10.28 Describe the means provided in the design of the scram discharge system and verify that it meets the criteria enumerated in the Generic Safety Evaluation Report BWR Scram Discharge System, dated December 1, 1980, and transmitted to you by NRC letter dated December 22, 1980.

RESPONSE

The criteria given in the Generic Safety Evaluation Report - BWR Scram Discharge System, dated December 1, 1980, are organized under the following headings: 1) functional criteria, 2) safety criteria, 3) operational criteria, 4) design criteria, and 5) surveillance criteria. The scram discharge system meets the criteria enumerated in the Generic Safety Evaluation Report - BWR Scram Discharge System, as described in the following discussions.

I. Functional Criteria

Functional Criterion 1:

The scram discharge volume shall have sufficient capacity to receive and contain water exhausted by a full reactor scram without adversely affecting control-rod-drive scram performance.

Compliance:

A minimum scram discharge volume of 3.34 gallons per drive is specified through the system design specifications. This minimum scram discharge volume is based on conservative assumptions as to the performance of the scram system. In the event of a coolant leak into the SDV, an automatic scram will occur before the required SDV available volume is threatened.

II. Safety Criteria

Safety Criterion 1:

No single active failure of a component or service function shall prevent a reactor scram, under the most degraded conditions that are operationally acceptable.

Compliance:

No single failure in the scram system design will prevent a reactor scram. The scram discharge system design meets the NRC acceptance criterion for Safety Criterion 1. Partial loss or full loss of service functions will result in either not adversely affecting the scram system function or a full reactor scram. The system requirements state that there shall be no reduction in the pipe size of the header piping going from the HCUs to and including the Scram Discharge Instrument Volume (SDIV). This hydraulic coupling permits operability of the scram level instrumentation prior to loss of system function. The scram level instrumentation will be to assure no single active failure prevents a reactor scram. See also the response to Question 10.29.

Safety Criterion 2:

No single active failure shall prevent uncontrolled loss of reactor coolant.

Compliance:

The current Grand Gulf design provides single vent and drain valves. To comply with the NRC Staff's position on this criterion, redundant vent and drain valves will be provided as part of the SDV modifications. The redundant SDV valve configuration assures that no single failure can result in an uncontrolled loss of reactor coolant. An additional solenoid operated pilot valve will control the redundant vent and drain valve. The vent and drain system will therefore be sufficiently redundant to avoid a failure to isolate the SDV due to solenoid failure. The redundant vent and drain valve's opening and closing sequences will be controlled to minimize excessive hydrodynamic forces.

Safety Criterion 3:

The scram discharge system instrumentation shall be designed to provide redundancy, to operate reliably under all conditions, and shall not be adversely affected by hydrodynamic forces or flow characteristics.

Compliance:

The current Grand Gulf design provides for redundant level sensing instrumentation. Water level in each scram discharge instrument volume (SDIV) is monitored by two level transmitters for the scram function. This configuration allows a one-out-of-two twice logic to initiate the automatic scram on high SDIV volume. An additional level transmitter monitors water level for the alarm and rod block function.

Modifications will be made to the existing design to add diverse instrumentation to the already redundant configuration. These modifications will be consistent with the NRC's alternative 1, as presented in the generic SER referenced in the question. These modifications will be implemented prior to the startup from the first regularly scheduled refueling outage. Instrument taps will be re-located from the vent and from the flow dynamics in the scram discharge system. This relocation of instrument taps will be accomplished prior to fuel load. The instrumentation arrangement incorporating the above described design modifications will assure the automatic scram function on high SDIV water level in the event of a single active or passive failure.

Safety Criterion 4:

System operating conditions which are required for scram shall be continuously monitored.

Compliance:

See response to Safety Criterion 3.

Safety Criterion 5:

Repair, replacement, adjustment, or surveillance of any system component shall not require the scram function to be bypassed.

Compliance:

The SDIV scram level instrumentation arrangement and trip logic allows instrument adjustment or surveillance without bypassing the scram function or directly causing a scram. Each level instrument can be individually isolated without bypassing the scram function. A one-out-of-two twice trip logic is employed.

III. Operational Criteria

Operational Criterion 1:

Level instrumentation shall be designed to be maintained, tested, or calibrated during plant operation without causing a scram.

Compliance:

See response to Safety Criterion 5.

Operational Criterion 2:

The system shall include sufficient supervisory instrumentation and alarms to permit surveillance of system operation.

Compliance:

Supervisory instrumentation and alarms such as accumulator trouble, scram valve air supply low pressure, and scram discharge volume not drained alarms, are adequate and permit surveillance of the scram system's readiness.

Operational Criterion 3:

The system shall be designed to minimize the exposure of operating personnel to radiation.

Compliance:

Minimizing the exposure of operating personnel to radiation is a consideration in equipment design and location.

Operational Criterion 4:

Vent paths shall be provided to assure adequate drainage in preparation for scram reset.

Compliance:

A vent line is provided as part of the scram discharge system to assure proper drainage in preparation for scram reset. Additional vent capability is provided by the vent line vacuum breakers.

Operational Criterion 5:

Vent and drain functions shall not be adversely affected by other system interfaces. The objective of this requirement is to preclude water backup in the scram instrument volume which could cause spurious scram.

Compliance:

The SDV vent and drain lines are dedicated lines that discharge into the suppression pool. Vacuum breakers on the SDV vent line and shut-off valves on the SDV vent and drain lines preclude water from siphoning back into the SDIV from the suppression. pool.

Design Criteria

Design Criterion 1:

The scram discharge headers shall be sized in accordance with GE OER-54 (Reference 20) and shall be hydraulically coupled to the instrumented volume(s) in a manner to permit operability of the scram level instrumentation prior to loss of system function. Each system shall be analyzed based on the plant-specific maximum in leakage to ensure that the system function is not lost prior to initiation of automatic scram. Maximum in leakage is the maximum flow rate through the scram discharge line without control-rod motion summed over all control rods. The analysis should show no need for vents or drains.

Compliance:

As discussed in response to Functional Criterion 1, a minimum scram discharge volume of 3.34 gallons per drive is specified through the system design specifications. Furthermore, the system requirements state that there shall be no reduction in the pipe size of the header piping going from the HCUs to and including the SDIV. The SDIV shall be directly connected to the scram discharge volume at the low point of the scram discharge header piping. These requirements satisfy the NRC's acceptance criteria for Design Criterion 1.

Design Criterion 2:

Level instrumentation shall be provided for automatic scram initiation while sufficient volume exists in the scram discharge volume.

Compliance:

See response to Functional Criterion 1 and Design Criterion 1.

Design Criterion 3:

Instrumentation taps shall be provided on the vertical instrument volume and not on the connected piping.

Compliance:

See response to Safety Criterion 3.

Design Criterion 4:

The scram instrumentation shall be capable of detecting water accumulation in the instrumented volume(s) assuming a single active failure in the instrumentation system or the plugging of an instrument line.

Compliance:

See response to Safety Criterion 3.

Design Criterion 5:

Structural and component design shall consider loads and conditions including those due to fluid dynamics, thermal expansion, internal pressure, seismic considerations and adverse environments.

Compliance:

The SDV and associated vent and drain piping is classified as ' portant to safety and required to meet the ASME Section III Class II a . Seismic Category I requirements. Design parameters such as temperature, pressure, and frequency, for limiting models of operation provide a design basis for supply of equipment as well as the interfacing piping design and analysis.

Design Criterion 6:

The power-operated vent and drain valves shall close under loss of air and/or electric power. Valve position indication shall be provided in the control room.

Compliance:

The present vent and drain valve design operation meets this criterion.

Design Criterion 7:

Any reductions in the system piping flow path shall be analyzed to assure system reliability and operability under all modes of operation.

Compliance:

See response to Design Criterion 1.

Design Criterion 8:

System piping geometry (i.e., pitch, line size, orientation) shall be such that the system drains continuously during normal plant operation

Compliance:

All SDV piping is required to be continuously sloped from its high point to its low point.

Design Criterion 9:

Instrumentation shall be provided to aid the operator in the detection of water accumulation in the instrumented volume(s) prior to scram initiation.

Compliance:

The present alarm and rod block instrumentation meets this criterion.

Design Criterion 10:

Vent and drain line valves shall be provided to contain the scram discharge water, with a single active failure and to minimize operational exposure.

Compliance:

See response to Safety Criterion 2.

IV. Surveillance Criteria

Surveillance Criterion 1:

Vent and drain valves shall be periodically tested.

Compliance:

Periodic testing of vent and drain valves are provided in Subsection 4.1.3.1 of the Grand Gulf Standard Technical Specifications (STS).

Surveillance Criterion 2:

Verifying level detection instrumentation when implemented shall be periodically tested in place.

Compliance:

Plant surveillance procedures associated with the Grand Gulf STS are currently under development and review. These procedures will provide for the verification that level instrumentation has been properly returned to service following testing.

Surveillance Criterion 3:

The operability of the entire system as an integrated whole shall be demonstrated periodically and during each operating cycle, by demonstrating scram instrument response and valve function at pressure and temperature at approximately 50% control-rod density.

Compliance:

STS subsection 4.1.3.1.4 provides periodic testing and acceptance criteria to demonstrate SCRAM discharge volume operability. This testing addresses the concern of this surveillance criterion.

References to the STS appearing in the above paragraphs are associated with the Proof and Review copy provided to MP&L dated August 4, 1981.

V. Implementation Summary

Design modifications, unless otherwise noted, will be completed prior to the startup from the first regularly sheduled refueling outage. These modifications primarily consist of the installation of a redundant set of vent and drain valves and of diverse water level instrumentation for the scram discharge instrument volume. As stated in the response to Safety Criterion 3, instrumentation tap relocations will be accomplished prior to fuel load. With the exception of these modifications, the current Grand Gulf design meets the criterion of the subject generic SER on BWR scram discharge systems.

Branch: Instrumentation and Controls Systems

Concern:

Provide additional information regarding water level instrument error expected from reference leg boil off during accident conditions.

Response:

1. Review of Reactor Water Level Measurement Instrumentation

The cold reference leg reactor water level measurement design for Grand Gulf is illustrated in Figure 1. Reactor vessel water level is measured by differential pressure transmitters which measure the difference in static head between two columns of water. One column is a "cold" (ambient temperature) reference leg outside the reactor vessel; the other is the reactor water inside the reactor vessel. The measured differential pressure is a function of reactor water level.

The cold reference leg is filled and maintained full of condensate by a condensing chamber at its top which continuously condenses reactor steam and drains excess condensate back to the reactor vessel through the upper level tap connection to the condensing chamber. The upper vessel level tap connection is located in the steam zone above the normal water level inside the vessel. Thus, the reference leg presents a constant reference static head of water to the high pressure tap on the d/p transmitter. The low-pressure tap of the transmitter is piped to a lower-level tap on the reactor vessel which is located in the water zone below the normal water level in the vessel. The low-pressure side of the transmitter thus senses the static head of water/steam inside the vessel above the lower vessel level tap. This head varies as a function of reactor water level above the tap and is the "variable leg" in the differential pressure measured by the transmitter. Lower taps for various instruments are located at various levels in the vessel water zone to accommodate both narrow- and wide-range level measurements (see Figure 2).

Typical reactor level indicators and recorders are shown on Figure 3. This figure also shows the condensing chamber elevations and set points elevations relative to the top of the active fuel.

2. Problem Description:

Small (e.g., .01 ft2) and intermediate (e.g., .04 ft2) break accidents (LOCA's) that discharge steam into the drywell (at temperatures as high as 340oF) for an extended time period result in substantial heatup of components/air in the drywell (including reactor water level sensing lines). If the reactor is subsequently depressurized below 118 psia, water in the reactor water level sensing lines located in the drywell will flash.

General Electric has conservatively evaluated many steam break accidents and his determined that, for the worst case scenario (small break accident with ADS operation after 1800 seconds), flashing will result in a loss of up to 20% of the water in the sensing lines. Water in the variable leg sensing line will be replenished by drain back from the reactor, while water in the reference leg sensing line will continue to be gradually depleted due to boil-off. If no operator action is taken, all of this water could, for the worst case, boil off after more than 10 hours after the accident. Loss of water from the reference leg results in a sensed reactor water level that is higher than the actual reactor water level. GRAND GULF reactor water level instrumentation utilizes FOUR reference legs for the narrow and wide-range level instrumentation. Utilizing this instrumentation, a level error of approximately 9' 6" could occur. It should be noted that all reactor water level activated safety trips will occur since they would initiate before the reactor is depressurized below 118 psia.

3. Operator Actions and Conditions that Prevent and/or Eliminate Flashing/Boil-Off:

Flashing/Boil-off will not occur if:

a) The break discharges two-phase fluid only;
b) The drywell achieves the higher temperatures AND level is recovered such that the saturated liquid spilling out of the break and cooling the steam lines and drywell environment terminates the heatup transient;
c) The reactor pressure is maintained above 118 psia.

In addition, even if flashing/boil-off were to occur, it would not be a concern if the operator follows the emergency procedure guidelines (EPG) and maintains reactor level in the normal water level range. Furthermore, the error due to flashing/boil-off will be eliminated if:

a) The operator follows the EPG and takes action to refill the reference leg after reactor depressurization if the temperature near the reference leg has exceeded the reactor saturation temperature and continues reactor injection until the temperature near the reference leg is below 212oF; or

b) The operator determines that a flashing/boil-off condition exists and takes corrective action to refill the reference leg. Indications available to the operator that indicate reference leg flashing/boil-off are:

1) erratic level indication
2) mismatch between narrow, wide and upset range level indicators and recorders (Note: Since EPG requires the operator to monitor water level from multiple indications, he should be aware of level instrument mismatch and hence flashing/boil-off conditions.)

4. Conclusion:

Considering the limited number of events, operator errors and conservative analysis assumptions described above, the probability of reference leg flashing/boil-off resulting in core uncovery is considered extremely low. Even if one assumes that the worst case scenario described above occurs, the operator would receive a level 2 alarm approximately 10 minutes prior to initial core uncovery.

Based on the above, it is concluded that the GRAND GULF reactor water level measurement instrumentation is acceptable.

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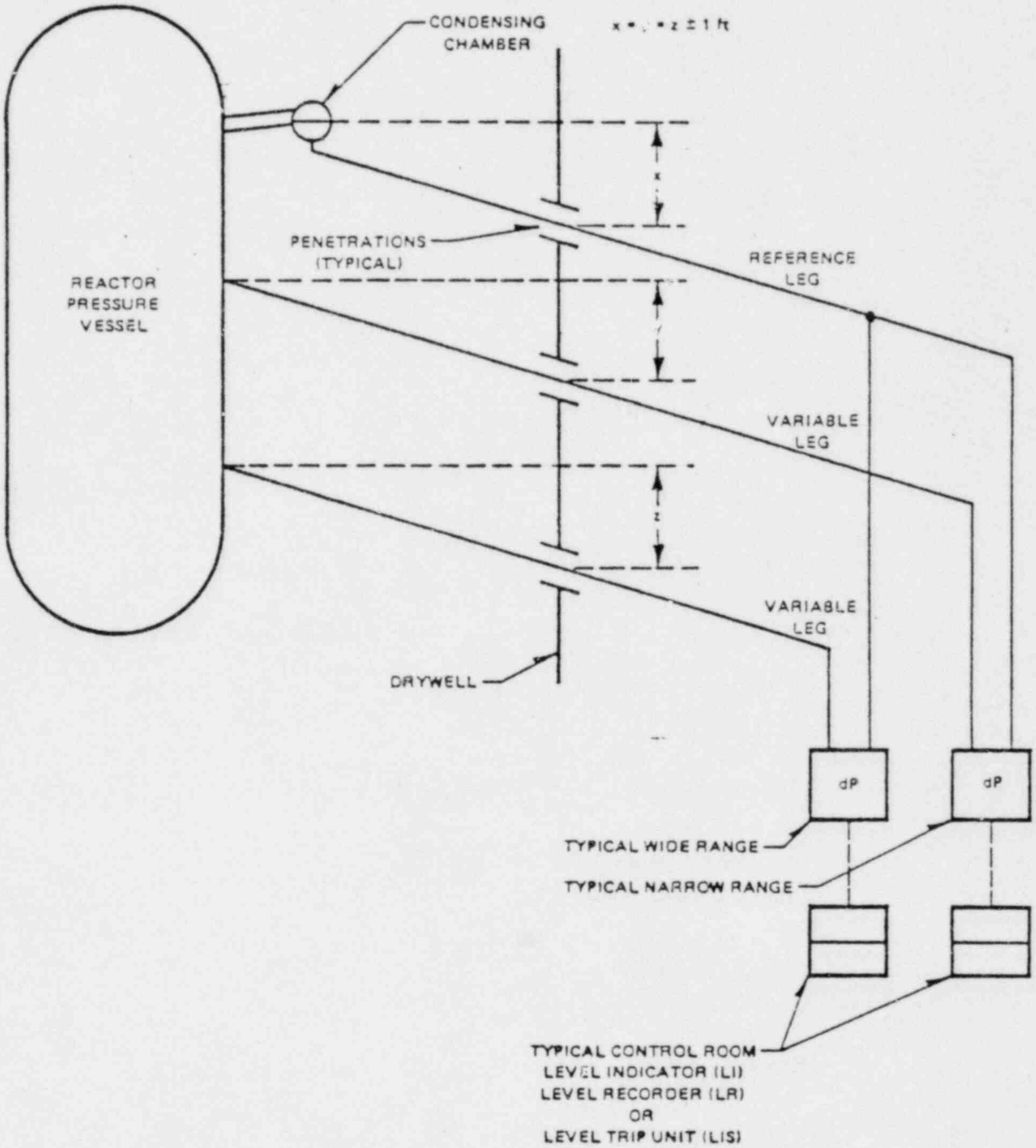

Figure 1 Cold Reference Leg Design

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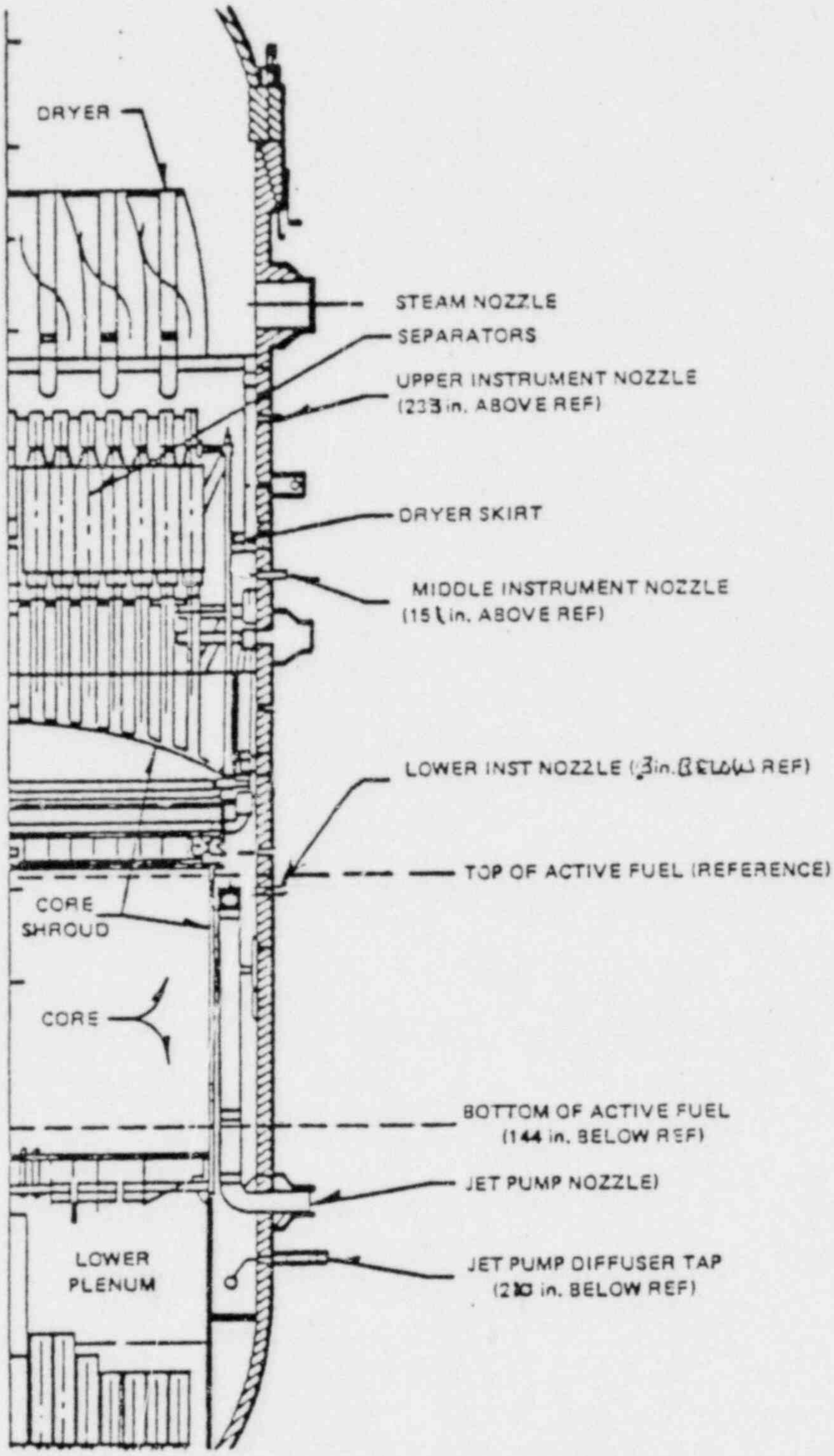

Figure 2 Location of Water Level Instrument Taps

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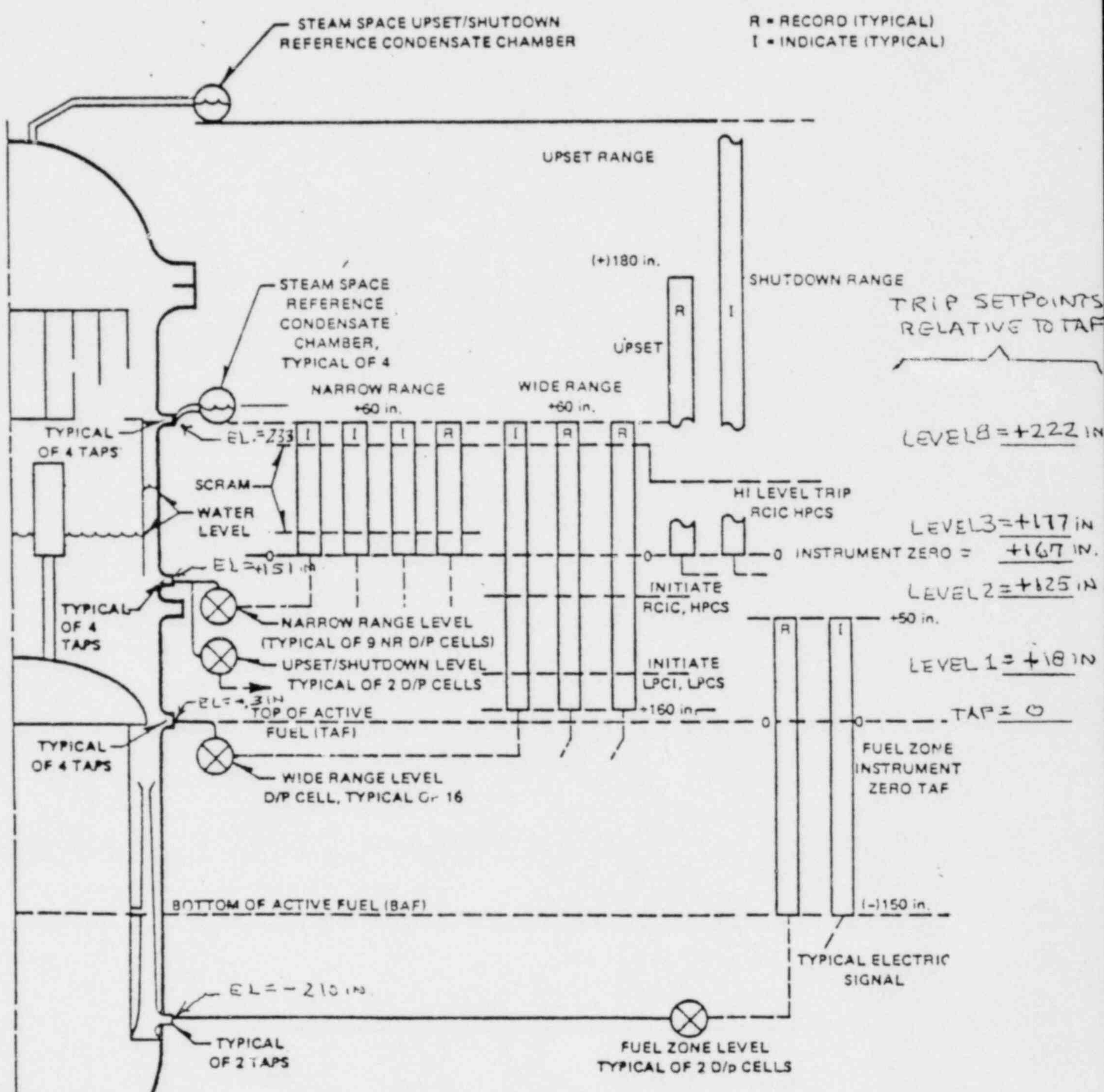

NOTES: 1. SEPARATE D/P CELLS ARE USED FOR NARROW RANGE INDICATION AND TRIP UNITS

2. INDICATION/RECORD AND TRIP UNITS FOR WIDE RANGE USE COMMON D/P CELLS

3. ALL ELEVATION VALUES ARE IN INCHES AND RELATIVE TO THE TOP OF ACTIVE FUEL (TAF

Figure 3 Reactor Level Indicators on Reactor Control Panels

Branch: Reactor Systems

Concern:

Discuss the effect of a realistic maximum FCV closing rate upon the DBA, indicate which single failure of ESF was taken and discuss single failure of ESF was taken and discuss the details of the analysis and results.

Response:

Failure modes and effects analysis have shown that, given a LOCA event, no single failure in the electronic/hydraulic controls can cause the FCV to close.

As a result of these considerations, FVC closure in the unbroken loop is not expected to occur during the LOCA event.

Even if the FCV were signalled to close for some unlikely reason (LOCA plus two failures: failure of drywell high pressure signal such that FCV lock-up does not occur, and failure of FCV controls), backup electronic velocity-limiters are included in the recirculation control system to limit FCV velocity to 10 $\pm$ 1% actuator stroke rate. Additional multiple specific component failures in these limiters must occur to cause full closure of the FCV at velocities in excess of this value. The combined probability of occurrence of these specific failure modes during LOCA is less than 106 per year. Accordingly, the electronically limited rate of 10 $\pm$ 1% of FCV actuator strike/rate is considered a realistic yet conservative closure rate.

FSAR Section 15.3.2 presents the results of dual failure of the recirculation FCV's which close at the maximum 11% per second stroking rate designed in as a constraint via the velocity limiter. Fuel centerline temperature and average surface heat flux decrease uniformly with time as in event 15.3.1 above, but at a slightly faster rate (-15%). MCPR remains essentially unchanged, and fuel does not fail.

Using approved standard licensing models, ECCS analyses were performed for a BWR/5 to determine the effect (sensitivity) on peak cladding temperature from FCV closure at the 11% per second rate. The calculated maximum peak temperature increase was 45oF.

Thus, the peak cladding temperature effect is concluded to be very small. The probability of FCV fast closure simultaneously with a LOCA is extremely remote. Accordingly, fast FCV closure in conjunction with the DBA-LOCA is not expected to occur and need not be compared to the maximum PCT criterion of 10 CFR 50.46.

The BWR/5 results are applicable to Grand Gulf, since the recirculation flow control systems are identical.

The attached Figure is a generic plot of the peak cladding temperature vs. time with and without FCV closure for the case which resulted in the largest PCT increase due to flow control valve closure. The initial increase in PCT is caused by the earlier loss of nucleate boiling with FCV closure. For time periods after boiling transition of the base case, the convective heat transfer coefficients for both cases are nearly identical for the remainder of the transient. This results in a higher heat removal rate for the FCV closure case and a decrease in the PCT difference to a value of 45oF at the time of reflooding. For plants which reflood later, the sensitivity to FCV closure will be smaller since the PCT difference decreases with time.

See Subsections 15.3.2 and 15.4.5 for analyses of FCV failures in the recirculation system.

Generic Figure

Comparison of Peak Cladding Temperature vs. Time for LOCA Analysis with and without Flow Control Valve Closure