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BOUCHEY,G.D. Washington Public Power Supply System
RECIP.NAME RECIPIENT AFFILIATION
SCHWENCER,A. Licensing Branch 2

SUBJECT: Forwards response to request for supporting documentation & per Tech Spec second draft discussed during 830523-26 meetings.Tentative schedule for preparation of proof & review copy listed.

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Washington Public Power Supply System
P.O. Box 968 3000 George Washington Way Richland, Washington 99352 (509) 372-5000

June 15, 1983
G02-83-522

Docket No. 50-397

Director of Nuclear Reactor Regulation
Attention: Mr. A. Schwencer, Chief
Licensing Branch No. 2
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Schwencer:

Subject: NUCLEAR PROJECT NO. 2
DRAFT TECHNICAL SPECIFICATION

During the meeting held between your Mr. D. Hoffman and our Messrs. M. R. Wuestefeld and P. Powell on May 23 through 26, 1983, various items were addressed requiring a Supply System response. Attached are those responses requested supporting documentation and other information pertaining to the WNP-2 Technical Specification second draft.

The meeting conducted in May was in response to a Supply System request to expedite the formation of a proof and review copy of our Technical Specifications. The expeditious response by the Commission is greatly appreciated and it is our belief that the meeting was very productive. A tentative schedule for preparation of the proof and review copy was discussed and is as follows:

6/10/83: Action item list closed out
6/17/83: WNP-2 Technical Specification into word processing
6/27/83: Proof and Review issued to Supply System and NRC internally for review
7/25/83: Review of Proof and Review copy completed
8/08/83: WNP-2 Technical Specification complete and ready for issuance with operating license

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The tentative nature of the schedule was discussed and was agreed to serve as an achievable goal.

For any additional information, please contact M. R. Wuestefeld (509-377-2501, x-2843) or P. Powell (x-2909).

Very truly yours,

G D Bouchey

G. D. Bouchey
Manager, Nuclear Safety and Regulatory Programs

MRW/tmh
Enclosure

cc: R Auluck - NRC
WS Chin - BPA
D Hoffman - NRC
A Toth - NRC Site

TECHNICAL SPECIFICATION LOG ITEM RESPONSES

Item #1: The turbine bypass valve response time definition is taken from GE Document 22A3805, Rev. 2, Turbine Generator and Steam Bypass System Design Specification. (Page #18 attached)

Item #2: The scram discharge volume alarm/rod block/scram capacities will be determined by a pre-op program test scheduled for June 20, 1983 and will be provided upon test completion. It is understood that this is for information only and that the content change to Item #8 on Page B 2-9 will be accepted.

Item #3: Given a stuck rod, LCO 3.1.3.1, Action a.1.c, requires compliance with surveillance 4.1.1.c or requires hot shutdown in next 12 hours. Action 3.1.3.1.a inclusively provides one hour to verify in-op rod separation, disarm the stuck rod mechanism, and comply with 4.1.1.c. One hour is sufficient to verify separation and disarm, but may not be sufficient time to analytically determine SDM. The computer code used, if performed using the on-site PRIME computer system, would take two to three hours to run. The Supply System has a contract with Boeing, Bellevue Division, to utilize their computer, which has been demonstrated successfully and takes approximately 100 seconds to determine SDM. The Boeing computer has usage limitations as do all computers owing to scheduled weekly maintenance and unexpected downtime. Our requested change was intended to provide the flexibility to initiate SDM determinations rather than complete it within an hour and verify adequate margin within six hours or be in at least HSD in the following 12 hours. Having the requirement to perform the analysis within one hour to determine present SDM and then provide six hours to reestablish SDM places more importance on performing the analysis than providing adequate SDM. It would appear that the one hour was selected to conform to the one hour action time in 3.1.3.1. a.1. It is our belief that this change does not represent a degradation of safety since we are not requesting an extension of the action statement time span, only the analysis time frame.

Item #4: LRG information provided to D. Hoffman during the May 23-25, 1983 meeting.

Item #5: Same response as Item #4 above.

Item #6: Alternate method is: "Performing an OD-7 on the process computer or ...".

Item #7: Table and graph originals will be sent by June 17, 1983.

Item #8: The reason the (j) note was removed for turbine throttle valve closure is because the actual relays used are separate from RPS. Since this distinction is not germane to the intent of the note, it should not be removed from the table.

Item #9: As discussed, the first stage main turbine pressure versus turbine power data will be gathered during the Power Ascension Test (Startup) Program. The only present source of data is a Westinghouse generated curve that indicates 180 psia at 25% of $14.298 \times 10^{+6}$ lbm/hr (Curve #AF104-0322).

Item #10: This item is to become a LRG-1 item, since it represents a generic issue.

Item #11: WNP-2 plant procedures are being written to perform the CFT for MSIV RPS logic on a monthly schedule and will cycle each MSIV using the independent test switch that enables slow closure testing to the approximately 93% open position to verify the RPS logic.

Item #12: The Supply System is presently negotiating with GE for the analysis to support the .07 MCPR penalty for loss of EOC-RPT. Our recommendation to the NRC is to leave the MCPR penalty values stated in LCO 3.2.3, Action a., open pending completion of the GE analysis.

Item #13: The vendor information for the triaxial time history accelerographs is ±1 g and was perceived to represent a 1 g range in two directions for each accelerometer. An 1 g range equally represents the range and can, therefore, be used in place of ±1 g.

Item #14: FSAR Section 2.3.3.1 specifies the instruments location at 33 and 245 feet.

Item #15: Regulatory Guide 1.68, Page 14, Paragraph 3, "Initial Criticality", requires a minimum count rate of at least one-half count per second on the startup channels before startup begins. WNP-2 is adopting this value for the purpose of providing operational flexibility. We have calculated that an additional margin of 156 days is provided with the .5 cps limit (not considering any intrinsic neutron sources, only the installed source). The same 2:1 signal to noise ratio applies and will be verified.

Item #16: FSAR Paragraph 6.4.4.2.1 states that the control room intake header automatically isolates the control room within 10 seconds in the event of a chlorine concentration of 5 ppm or greater. The vendor preset the Model 7040 F.A.N. units to signal an alarm in eight seconds at 5 ppm Cl_2 gas concentration and provides "within eight seconds" as the alarm response time test acceptance criteria. Ten seconds was selected to conform to the FSAR.

Item #17: Reactor Vessel Overpressure Protection Design Report, Document 22A4106, Rev. 2, states that a MSIV closure with scram signal derived from indirect means (flux versus directly from MSIV closure logic) is the most severe overpressure transient. The analyses results, indicated on Figure 3 and FSAR Figure 5.2-4, indicates that 12 operating safety relief valves are required to limit the pressure response to less than the code limit of 1375 psig. A copy of the document is attached for your information. Note that Susquehanna Tech Spec requires 10 of 16 valves operable.

In addition, the Reactor Vessel Overpressure Protection Analysis was redone using the ODYN code. This analysis shows a 28 psi improvement (reduction) in peak vessel pressure assuming a MSIV-flux scram initiating event. Using standard GE design practice of applying a 40 psi design margin to code pressure limits results in a peak pressure analytical limit of 1335 psig. With the 28 psi improvement from ODYN, the analytical pressure transient as a function of safety relief valve capacity intersects the analytical limit at 11+ valves required operable. Therefore, 12 SRVs was selected as the minimum operable number of valves.

Note on history of 18 relief valves on WNP-2: WNP-2 is a BWR-5 1969(?) product line plant which initially had prompt relief trip (PRT) and relief valve augmented bypass (REVAB) logic in its design. This logic allowed the NSSS to ride out gross disturbances from the BOP (e.g., turbine trip and load rejections @ 100% steam flow) without a reactor scram and attentive loss of capacity factor/plant availability upon scram recovery by passing the steam flow to the suppression pool whenever steam flow to the turbine was interrupted. Eighteen (18) valves were required to allow the NSSS to ride out the pressure transient. Without PRT/REVAB, the number of valves required is determined solely by the overpressurization analysis discussed above.

Item #18: The WNP-2 Plant Design Assessment Report for SRV and LOCA Loads (included in Amendment #23 to our FSAR) analyzed containment loads given SRV actuation. The loads represented in "SRV Loads - Improved Definition and Application Methodology for Mark II Containments" technical report has been transmitted and approved by the Commission (letter G02-80-172, dated August 8, 1980). In essence, due to the analyzed load levels for WNP-2 containment, the low-low set function is not required.

Item #19: The actual alarm setpoints in Table 3.4.3.2-2 are intended to alert the operator to high pressure system leakage into a low pressure system. The setpoints will be included in the WNP-2 calibration procedures. To preclude impacting the Tech Spec review process, it is requested that the actual setpoints be deleted as they are presently not available. Their omission will not preclude either the CFT or CC surveillance tests required in 4.4.3.2.3.

Item #20: We have been unable to substantiate the industry-wide usage of using a 30-minute delay time for evaluating the activity levels at the offgas pretreatment radiation monitor. Since the values presented only effect the sample frequency, delete the 30-minute delay mixture.

Item #21: The Supply System has been unable to determine the bases for the three EFPY limit mentioned in the SER. The curve presently in the FSAR is being amended to comply with the latest GE supplied information and is the curve presented in the second draft of our tech spec. We are continuing to search for the bases for the statement in the SER through communications with your Materials Research and Materials Engineering Branches. Any information concerning the source and technical justification for the three EFPY limit would be appreciated.

Item #22: For the purposes of Tech Specs, the GE design bases values for required pump capacities will be used.

HPCS: Flow = 6350 gpm at $\geq$ 200 psid (between reactor & suction source)
LPCS: Flow = 6350 gpm at $\geq$ 128 psid (between reactor & suppression pool)
RHR (LPCI): 7450 gpm at $\geq$ 26 psid (between reactor & suppression pool)

The values from the pre-op test results will be reflected in the WNP-2 Pump and Valve Testing Program (ASME Section XI).

Item #23: FSAR Section 9.3.1.2.2 specifies the 30-day capacity for the nitrogen backup system for the ADS function of the SRVs to be 257 scf using commercially available bottles pressurized at 2490 psig. In the event the actual bottles purchased are not identical to those specified (but will contain as a minimum 257 scf), the pressure may vary. This variance was accounted for in the manner the LCO surveillance was previously presented. Stipulating a pressure of 2490 psig may cause a change later, but is consistent with our FSAR. Any change will be immediately forwarded to you prior to license issuance, if the form presented is not accepted.

Item #24: Suppression pool water level will be verified on a 12-hour frequency.

Item #25: FSAR Section 6.7.2.3 states the MSIV LCS blower capacity is 50 cfm at about -20 inches H_2O suction pressure. Larger MSIV leakage rates can be accommodated at higher blower flow rates, therefore, the WNP-2 flow rate is conservative when compared to LaSalle.

Item #26: LCO 3.6.1.8 should provide unrestricted usage of the 24 and 30 inch purge valves during reactor power operation less than 15%. This power level and philosophy is consistent with LCO 3.6.6.2, Drywell and Suppression Pool Oxygen Concentration. In addition, given the environmental qualification level and analysis to support the 24 and 30 inch valves' capability to operate during a DBA LOCA, locking or seal closure of the valves is unnecessary while above 15% power. Having to purge the containment before reaching Mode 3 precludes drywell entry at sufficiently low power levels to locate and repair any leaks. The six inch valves would then be used exclusively to purge when above 15% power.

The techniques employed in the WNP-2 10CFR50, Appendix J, leakage testing applies a single valve leakage rate limit for multiple valve testing. This conservative approach should be acceptable to the CSB.

Item #27: The operability range of the suppression pool is based on 1) minimum level requirements in the event of a DBA LOCA to provide adequate pressure suppression for maintaining primary containment integrity and 2) to limit downcomer submergence and provide sufficient suppression pool free air volume. FSAR Table 6.2-1 gives various containment design parameters. Among them are the 12 foot submergence limit and the minimum capacity of 112,197 $ft.^3$, plus 15,000 $ft.^3$ below the downcomers. The problem in conforming to the present LCO 3.6.2.1 format is that the suppression pool capacity at a level corresponding to 12 feet submergence is not known nor necessarily important, and the suppression pool capacity is indicated in level not volume. Therefore, to provide the operator with useful data, the LCO should provide the following:

Maximum level of 466' 4-3/4" elev., corresponding to a maximum downcomer submergence of 12 feet, and minimum level of 466' 0-3/4" elev., corresponding to a suppression pool water volume of at least 127,197 $ft.^3$.

Item #28: No Supply System response required.

Item #29: The spray pond inspection for sediment depth mentioned in SER Section 9.2.5 will be covered by a procedure in the WNP-2 Scheduled Maintenance System (SMS) and is not considered Tech Spec related.

Item #30: This item to be pursued by LRG-1.

Item #31: This item covered in Item #1 above.

Item #32: No Supply System response required.

Item #33: The GE design for HPCS auto starts on other than LOCA signals has a three second time delay to allow sufficient time for SM-2 to pickup the HPCS loads. Given a LOCA signal, the diesel auto starts and the 10 second starting time applies. The LCO is correct as presented with respect to diesel starting times.

Item #34: Voltage indication does not exist on all of the power supplies listed in LCOs 3.8.3.1 and 3.8.3.2. The WNP-2 plant procedures verify power distribution using a combination of voltage verifications and correct breaker alignment. The format presented in the Tech Spec is acceptable provided it is understood that not all MCCs and power panels listed have the means to verify voltage by indication either locally or in the control room. The Supply System
has committed to adding selected power indications per our response to Regulatory Guide 1.97.

Item #35: LCO 3.8.4.2 surveillance requirement 4.8.4.2.a.1 and .2 should remain as presented in the standard. WNP-2 will commit to testing 10% of the listed 6.9 kV and 480 V breakers every 18 months. For the 4.8.4.2.a.3 surveillance requirement of testing 10% of the fuses listed, WNP-2 will commit to replacement versus testing as was presented in our submittal.

Item #36: The revised table to include the secondary protective devices in Table 3.8.4.2-1 is attached. The items marked later will be provided by June 27, 1983.

Item #37: The emergency diesel fuel tank capacities listed in the FSAR and SER are nominal values. The minimum capacities listed in LCO 3.8.1.1 are based on providing sufficient fuel for seven days of rated capacity operation per Regulatory Guide 1.137. Full load consumption rate for Diesel Generator #3 has been demonstrated to be approximately 3.28 gpm.

Item #38: FSAR Section 15.7.4.5.1 stipulates Regulatory Guide 1.25 and SRP 15.7.4 as the bases for the design analysis used to evaluate fission product release during a fuel handling accident. The value for water depth assumed by GE in the analysis for 1) above the reactor flange and 2) above top of irradiated fuel was 22 feet. LCOs 3.9.8 and 3.9.9 should state 22 feet as the minimum depth required.

Item #39: The paragraph in the bases section for LCO 3/4 2.3, MCPR, discusses the K_f curve. Norma Stier, GE-San Jose, notified the Supply System via their WNP-2 Tech Spec review process that the paragraph applies to BWR-4 plants employing motor generator sets. WNP-2 reactor recirculation system uses a FCV for recirculation flow control and is a BWR-5. The K_f bases for WNP-2 is adequately defined in the previous two paragraphs, therefore, delete the paragraph as presented.

Item #40: No Supply System response required.

Item #41: Graph on Page B 3/4 4-7 will be provided by June 17, 1983.

Item #42: Amendment #30 to the WNP-2 FSAR discusses the on-site and off-site Safety Engineering Group, presently referred to as the Nuclear Safety Assurance Group (NSAG). A copy of the amendment is attached. Therefore, Page 6-7 is correct as presented.

Item #43: Amendment #31 to the WNP-2 FSAR has revised the status of the Administrative Manager and removed the non-voting description. Our plant procedures give the Plant Manager approval authority and does not provide for member voting. Each member participates, but approval or disapproval responsibility is delegated to the Plant Manager exclusively; therefore, the voting stature does not apply at WNP-2.

Item #44: WNP-2 FSAR Chapter 13 has been recently revised and germane sections of the SER should be reviewed for compliance. With respect to the organization charts submitted in the Tech Spec, an organization chart in Appendix B is attached for your information and to illustrate Tech Spec submittal compliance.

Item #45: The Corporate Nuclear Safety Review Board (CNSRB) does not direct the audit function described in 6.5.3. The CNSRB is a committee having no organizational responsibilities. At WNP-2, the Director of Licensing and Assurance is responsible for the direction of scheduling and ensuring the audit function is completed. Therefore, Section 6.5.3 should list the Director of Licensing and Assurance versus the Audits Manager as directing the audit function.

Items #46 and #47: The changes presented to the Commission within Draft #2 in 6.10.3 and 6.10.3.i may be omitted per your request. Sufficient detail is provided in the OQAM to provide for the concerns presented.

Item #48: Our proposed change was intended to accurately reflect the logic employed in the permissive logic that allows operation of the condenser low vacuum keylock bypass. The logic uses limit switches on the stop valves at the 90% open position. The valves are not throttle type and are, therefore, full open or closed (except during turbine roll which would require a main condenser vacuum). The paragraph's present form addresses the situation adequately, so can be retained if desired. The extent of "for information only" data is not often readily discernable.

Items #49 and #50: The relatively large differentials between the trip point and allowables for leakage detection type signals are based on drift allowance (see documentation attached, Step #11). As we discussed, the concern was whether or not Regulatory Guide 1.105 was used in the determination of these values. A typical B&R work sheet is provided to illustrate compliance to the Regulatory Guide and was provided by B&R for that purpose.

Item #51: The setpoint of 770 gpm and allowable of 900 gpm was confirmed as correct by GE. The bases for these values is the concern with limiting the amount of LPCS flow that is bypassed versus providing minimum flow protection. The LPCS mini-flow valve is normally open in the standby mode.

Item #52: The new fuel criticality monitor setpoint at WNP-2 is 10 R/hr. The calculation was performed in compliance with 10CFR70.24.a.1. The spent fuel storage pool monitor was listed in error as a criticality monitor and should be deleted from Table 3.3.7.1-1 per 10CFR70.24.a.

Items #53, #54, and #55: No Supply System response required.

Item #56: SSER 2, Page 6-3, discusses the Supply System's response to TMI Item II.K.3.28. The response is noted as acceptable to the NRC and does not require leakage testing of the ADS accumulators as was previously recommended. However, the description does require 18-month calibration of the bottle pressure gages which was never committed to by the Supply System. This appears to be an addition without prior notification to the Supply System. We are in the process of preparing a letter to take exception to the requirement and will be issued by our Licensing Department. The other three requirements are already in the WNP-2 Tech Spec.

Additional Information

1. Administrative Section 6.0, Paragraph 6.2.3.1, should be changed to have the NSAG make recommendations to the Plant Manager instead of the Director of Safety and Security. Our plant procedures presently stipulate that the original document be sent to the Plant Manager with a copy sent to the Operational Assurance Programs Manager.

2. Administrative Section 6.0, Paragraph 6.12.2, should refer to Shift Manager instead of Shift Foreman. This is an error in the second draft submittal.

3. Administrative Section 6.0, Paragraphs 6.13.2.6, 6.14.2.a.3, and 6.14.2.b should refer to Plant Manager instead of the On-Site Review and Investigative Function. This is an error in the second draft submittal.

4. LCO 3.3.2, Table 3.3.2-2, Item 5.d, RCIC Steam Supply Pressure - Low Trip Setpoint, should be 62 with an allowable of 58 psig. The setpoint should be the same as Item 4.c. This is an error in the second draft submittal.

5. The undervoltage trip point and allowable information provided in Table 3.3.3-2, Item D.1, is correct for Division 1 and 2 only (SM-7 and SM-8). Division 3 (SM-4) setpoint and allowable data is presented within the new format for Item D.1 as follows:

[illegible]

D. Loss of Power

Trip Setpoint

1. 4.16 KV Emergency Bus Undervoltage (loss of voltage##)

   a. Division 1 and 2

      a. Same values previously presented
      b. Same values previously presented

   b. Division 3

      a. 4.16 KV Basis - 3016 + 90 Volts
         120 V Basis - 87 ± 2.5 Volts

Allowable Value (for Division 3 only)

a. 3016 ± 180 volts
b. 87 ± 5 Volts

The values presented are higher in magnitude than Divisions 1 and 2 and are, therefore, conservative by comparison.

6. GE has recently presented the Supply System with some recommended changes to the electrical distribution LCOs. They are being reviewed and will be presented to you by June 17, 1983.

7. A recent conversation with your Mr. Joel Page and our Mr. T. F. Hoyle discussed the completion of an EG&G Report on intersystem leakage. The report recommends a valve leakage limit that is contrary to those presently in LCO 3.4.3.2.d. The limit presented is 1/2 gpm per inch valve diameter (nominal valve size assumed) up to a limit of 5 gpm per valve. We request your pursuit of utilizing these new values and offer our assistance if required.

2.2 (continued)

Item # 1

Upon a turbine trip or generator load rejection, the start of the bypass valve flow shall be delayed no more than 0.1 sec after the start of the stop or control valve closure, and at least 80 percent of the capacity of the bypass system shall be established within 0.3 sec after the start of the stop or control valve closure.

Signals shall be provided to sense bypass valve failure when the bypass is required to open if a full load rejection without a reactor scram is required. These signals shall be provided from four physically separated DPST position switches(1) divided equally among the bypass valves. In the case of three bypass valves being provided, two switches shall be provided per valve (six switches total) The switches shall be open when the bypass valves are closed, and shall not be subjected to spurious closure under normal full power load oscillations. (See bypass valve bias requirements, page 5).

The limit switch contacts shall close within 10 milliseconds after the bypass valves on which the switches are mounted have opened more than 10 percent of their stroke. The switches shall be wired in accordance with Paragraph 2.3.

In addition to the above contact requirements, a standard (Non IEEE 279 Isolated) contact closure of the form double pole single throw shall be provided from the steam bypass valve control relay system, or from a position switch on the "first to operate" steam bypass valves. This contact closure is required for use in the Nuclear Steam Supply System computer for logging functions and turbine steam flow measurement correction, in event of turbine bypass valves opening.

Controls shall be designed so that the valves will close on loss of control system electric power or hydraulic system pressure.

The Customer/AE shall provide four independent main condenser vacuum switches for the purpose of providing an isolation signal to the NSSS main steam isolation valves. Each vacuum switch must have its own isolation (root valves) and pressurizing source connection for testing. The wiring and separation requirements for these switches shall be in accordance with IEEE 279. Pressure switches shall have SPST contacts,(1) contacts to open on low vacuum. The vacuum switch setting shall be determined by the Customer/AE such that it is compatible with safe turbine and main condenser operating and design conditions should loss of vacuum occur.

(1) Contacts operate a 115V-ac 50/60 Hz trip coil circuit - Burden 32VA, 12 watt (holding).

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e) If this is not the case, check the calibration of the instrument with the calibration card. If the instrument indicates the correct calibration with the card, refer to the manufacturer.

f) If a calibration check is to be performed on the instrument only, disconnect the inlet line from the bulkhead fitting below the alarm reset switch.

g) Assure the inlet flow is 250 cc/min and allow the instrument to sample from a reservoir supplied at 300 cc/min.

h) Note the meter reading after 3 minutes. This should be equal to the gas concentration $\pm$20%.

i) Recheck the zero setting after removal of the concentration gas (allowing 5 minutes for the instrument to return to zero). Reset if necessary.

j) Refit the sample line, etc.

5.3 Alarm Time Response Test Procedure

a) Set the instrument flow at the upper red line.

b) In order to test the alarm time, a calibrated gas mixture of 5 ppm. must be supplied to the inlet of the 10 ft. sample line at a flow of 12 lpm.

c) Time the response of the instrument from the introduction of the sample to the activation of the alarm.

d) The alarm should actuate within 8 seconds.

INSTRUCTIONS FOR INSTALLING AND ADJUSTING NEW OR REPLACEMENT ELECTRONICS MODULE

1. Remove 7000 Series Chassis from blue case.
2. Insert electronics module and fresh unexposed tape into the gate.
3. Connect to power cord and turn on instrument. Use extreme caution. Do not touch any exposed terminals, wires or contacts.
4. Adjust Zero pot on the module so the front panel meter reads zero ppm.
5. Activate test switch: instrument should indicate full scale deflection on panel meter. If it doesn't, with test switch turned on, adjust pot VR2 on module until meter reads full scale.
6. Recheck zero and adjust if necessary. Repeat 5 and 6 until no further adjustments are needed.

Calibration Procedure with Test Jack

The test jack located on the front of the instrument is connected in series with the analogue output; the correct output is verified by connecting an ammeter to the test jack. A zero ppm meter reading is 4ma $\pm$ 10% and a full scale (4ppm) is 20ma$\pm$10%.

7. Insert test calibration card in calibrator position and adjust calibration pot as necessary to get proper reading.
8. Recheck zero and adjust as necessary.
9. Repeat steps 7 and 8 until no adjustment is necessary. Your instrument is now calibrated and ready for operation.

Item # 16

2.1 Zero Indication

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Item # 16

During normal operation, the meter indication may vary ± 5% from zero. This is due to the surface texture of the tape and provides a positive check on tape movement and photocell sensitivity.

2.2 Alarm Operation.

The alarms will be initiated whenever the gas concentration reaches or exceeds the SET ALARM point. Hysteresis is designed into the instrument such that, as the gas concentration falls below the SET ALARM point, there is a delay in terms of gas concentration before the alarm circuit is de-activated. This feature is to ensure that slight variations in gas concentration around the alarm level will not result in the intermittent operation of the alarm.

2.3 Response Time

Because of a delay between the gas being sampled and the stain being monitored, a delay in the alarm response occurs. This delay will depend on the alarm setting, gas concentration, and input flowrate.

Note: Model 7040 F.A.N. units are set to signal an alarm in 8 seconds at a 5 ppm Cl_2 gas concentration.

Should it be necessary to put the unit into immediate service elsewhere, wind the take-up spool by hand until a clear portion of tape is present at the detector head.

2.4 Tape Movement Indicator

The red-colored indicator is a mechanically-operated device situated adjacent to the detector head. Its function is to provide an indication of tape movement by deflection of a pointer.

2.5 Test Switch - Test-Jack

Operation of the test switch will cause the concentration meter to register a full scale deflection and will activate the alarm. The test-jack allows an external meter to check the 4-20 MA output.

2.6 Alarm Reset Swtich

The Alarm Lamp will remain on until reset with this switch. The signal level must be below the alarm set point for the alarm to reset.

2.7 Flow Indicator Lamp

Two pressure switches are wired in series with the flow lamp. These switches assure the proper flow conditions for the correct operation of the instrument. The optics block inlet is maintained at a 1 in. water gage positive pressure and the Rena diaphragm pump maintain a 6 in. (WG) negative pressure. In the event of failure of either the Rena or Thomas Pump, the pressure switch would deactivate the Master Fault Relay and signal an alarm.

2.8 Master Fault Lamp

A master fault will be indicated by an extinguished lamp in one of the following conditions: (a) A low signal output below zero reading
(b) Loss of power AC or DC

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6.4.4.2 Toxic Gas Protection

6.4.4.2.1 Chlorine

The only toxic chemical stored on site in substantial quantities is chlorine which is stored in cylinders as a liquid at three locations as shown in Figure 6.4-2. The largest amount of chlorine stored is used for circulating water chemical treatment. For this service as many as 10 two thousand pound cylinders of chlorine are stored at one time in the chlorination building. The chlorination building is located approximately 280 meters from the control room. The 10 chlorine cylinders are valved into a manifold for chemical feed to the circulating water system. No more than one cylinder (2000 pounds) is opened to the manifold at any one time.

The protection afforded control room operators from an on site chlorine release conforms to the requirements of Regulatory Guide 1.95 (Rev. 1, January 1977) for a "Type III" control room to the extent described below:

a. Redundant quick response chlorine detectors are provided in the control room intake header which automaticaly isolate the control room within 10 seconds in the event of the detection of a chlorine concentration of 5 ppm or greater.

b. The normal fresh air intake rate is 1000 cfm which represents a rate of approximately 0.29 volume changes per hour as compared to the maximum allowable rate of 0.3 air changes per hour.

c. The calculated infiltration rate into the control room at 1/8-inch water gauge pressure differential is approximately 0.022 air exchanges per hour as compared to the allowable rate of 0.06 air exchanges per hour for "Type III" control room. Chlorine concentration calculations do not take credit for an infiltration rate less than 0.06 air changes per hour.

d. The maximum quantity of chlorine that is released due to a single failure is 2000 pounds 280 meters from the control room. Interpolation of the quantities listed in Table 1 of Regulatory Guide 1.95 (Rev. 1) indicates a release of up to 12,000 pounds of chlorine at 280 meters is acceptable for a Type III control room.

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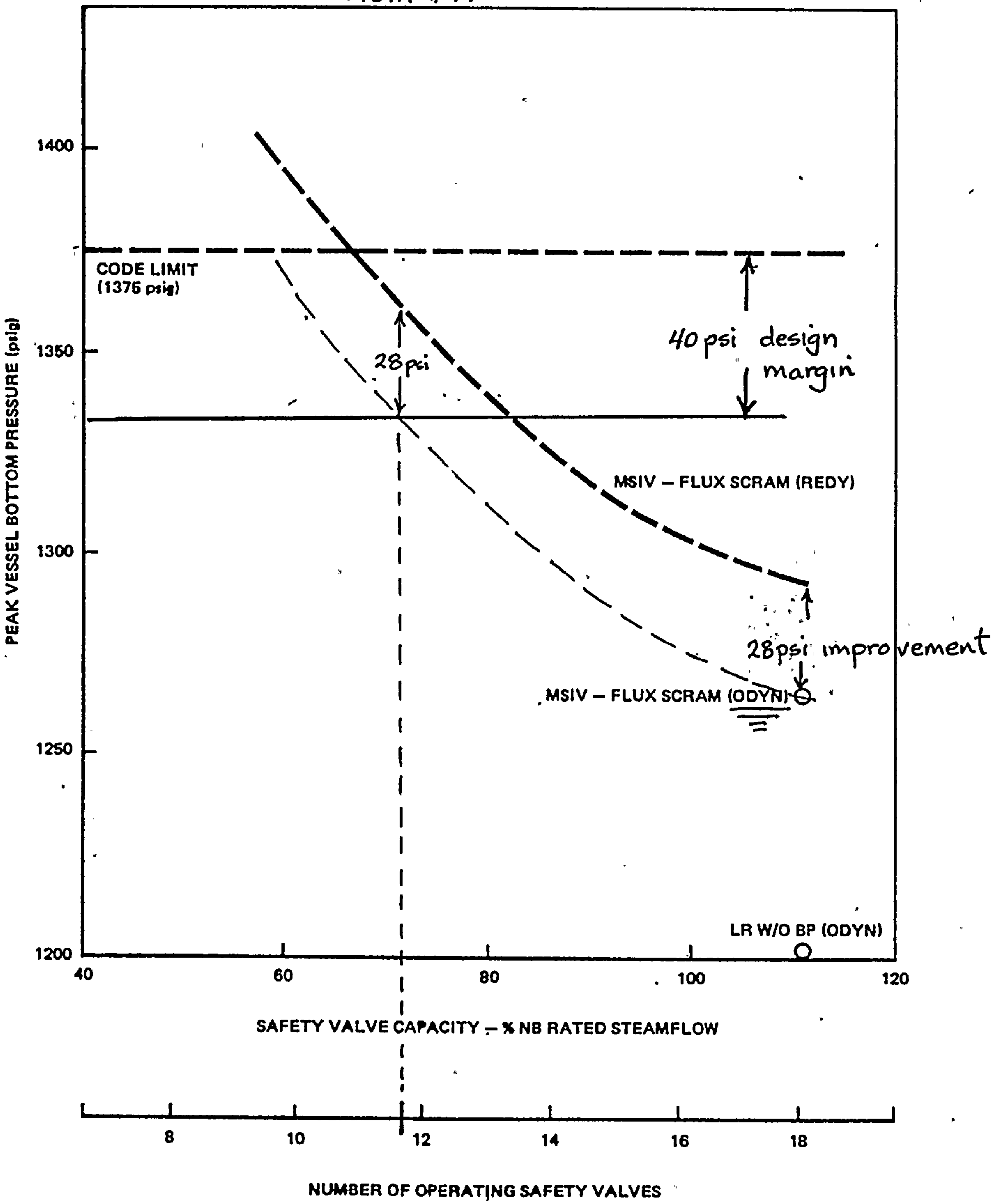
Item #17



WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	PEAK VESSEL PRESSURE VERSUS SAFETY VALVE CAPACITY	FIGURE 5.2-4

EIS Ident: RV OVPRESS PROT

GENERAL ELECTRIC REVISION STATUS SHEET

NUCLEAR ENERGY DIVISION

Item #17

22A4106

CONT ON SHEET 2 SH NO. 1

DOCUMENT TITLE REACTOR VESSEL OVERPRESSURE PROTECTION

☐ SPECIFICATION ☐ DRAWING ☒ OTHER

TYPE DESIGN REPORT

FMF NUCLEAR BOILER SYSTEM

MPL No. B22-5030

LEGEND OR DESCRIPTION OF GROUPS

HANFORD 2

RECEIVED
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		NH03531 2/8/80 CHKD BY: J Garcia K C Yee 2/1/80 J GARCIA K.C.YEE	

BRM FILE NUMBER 2 48 1401

2-(1970)

Washington Public Power Supply Sys
WPPSS Nuclear Project No 2
W. O. 2808

BURNS AND ROE, INC.
ORADELL, N.J.-HEMPSTEAD, N.Y.
-LOS ANGELES, CAL

REVIEWED AS CHECKED BELOW
☒ Approved for Fabrication............A
☐ NOT Approved..........................NA
☐ Approval as noted for Fabrication....AN
See Acceptance Note-Para. 1.2.
of Appendix A, Section 1 B
☐ For Information Only......................I

Subject To All Contractual Provisions
This Review does not imply acceptance of any material or equipment not fulfilling all specification requirements.

PROCESSED BY DATE
K.S./P.G. 6/13/80

PRINTS TO					
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CHKD BY ES HUBBARD E S Hubbard 3-23-77	DG ROBERTS MAR 29 1977	CONT ON SHEET 2	SH NO. 1

Abstract

This report provides sufficient information and documentation to show compliance with all requirements of Article NB-7000 of ASME Pressure Vessel Code - Section III, 1971, Power Plant Components (up to and including Summer 1971 Addenda), in the area of the vessel overpressure protection design of the Hanford 2 nuclear pressure vessel. The effects on the vessel pressure transients of valve capacity are also shown.

Table of Contents

List of Illustrations

1. INTRODUCTION

1.1 The vessel overpressure protection system is designed to satisfy the requirements of Section III, Nuclear Power Plant Components, of the ASME Boiler and Pressure Vessel Code. The general requirements for protection against overpressure as given in Article NB-7000 of Section III of the Code recognize that reactor vessel overpressure protection is one function of the reactor protective systems and allows the integration of pressure relief devices with the protective systems of the nuclear reactor. Hence, credit is taken for the SCRAM protective system as a complementary pressure-protection device. The NRC has also adopted the ASME Codes as part of their requirements in the Code of Federal Regulations (10CFR50.55A).

2. DESIGN BASIS

2.1 Safety Valve Capacity. The safety valve capacity of the Hanford 2 plant is adequate to limit the primary system pressure, including transients, to the requirements of the ASME Boiler and Pressure Vessel Code, Section III, 1971, Nuclear Power Plant Components (up to and including Summer 1971 Addenda). The essential ASME requirements met by this analysis are:

2.1.1 It is recognized that the protection of vessels in a nuclear power plant is dependent upon many protective systems to relieve or terminate pressure transients. Installation of pressure relieving devices may not independently provide complete protection.

2.1.2 The safety valve sizing evaluation assumes credit for operation of the SCRAM protective system which may be tripped by any one of three sources; i.e., a direct, flux, or pressure signal. The direct SCRAM signal is derived from position switches mounted on the main steamline isolation valves, or the turbine stop valves, or from pressure switches mounted on the dump valve of the turbine control valve hydraulic actuation system. The position switches are actuated when the respective valves are closing, and following 10-percent travel of full stroke. The pressure switches are actuated when a fast closure of the control valves is initiated. Further, no credit is taken for power operation of the pressure relieving devices. Credit is taken for the dual purpose safety/relief valves in their ASME Code qualified mode of safety operation.

2.1.3 The nominal pressure setting of at least one safety/relief valve connected to any vessel or system shall not be greater than a pressure at the safety/relief valves corresponding to the design pressure (1250 psig) anywhere in the protected vessel.

2.1.4 The rated capacity of the pressure relieving devices shall be sufficient to prevent a rise in pressure within the protected vessel of more than 110 percent of the design pressure (1.10X1250 psig = 1375 psig) for events defined in Paragraph 4.3.1.

2.1.5 Full account is taken of the pressure drop on both the inlet and discharge sides of the valves. All combination safety/relief valves discharge into the suppression pool through a discharge pipe from each valve, which is designed to achieve sonic flow conditions through the valve; thus providing flow independence to discharge piping losses.

3. METHOD OF ANALYSIS

3.1 To design the pressure protection for the nuclear boiler system, extensive analytical models representing all essential dynamic characteristics of the system are simulated on a large digital computing facility. These models include the hydrodynamics of the flow loop, the reactor kinetics, the thermal characteristics of the fuel and its transfer of heat to the coolant, and all the principal controller features (such as, feedwater flow, recirculation flow, reactor water level, pressure, and load demand). These are represented with all their principal non-linear features in models that have evolved through extensive experience and favorable comparison of analysis with actual BWR test data.

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3.1.1 A detailed description of this model is documented in licensing topical report NEDO-10802, Analytical Methods of Plant Transient Evaluations for the GE-BWR, R.B. Linford. Included within this model are components of the reactor vessel pressure protection system, which system is the subject of this report. Dual relief/safety valves are simulated in the non-linear representation, and the model thereby allows full investigation of the various valve response times, valve capacities, and actuation setpoints that are available in applicable hardware systems.

3.1.2 Typical capacity characteristics as modeled are represented in Figure 1 for the relief/safety valves. The associated bypass, turbine control valve, and mainsteam isolation valve characteristics are, of course, also represented fully in the model.

4. SYSTEM DESIGN

4.1 A parametric study was conducted to determine the required steam flow capacity of the safety/relief valves based on the following assumptions.

4.2 Operating Conditions

4.2.1	Operating Power	-	3464MWt (104.25 percent of reactor rated power)
	Vessel Dome Pressure	-	1020 psig
	Steamflow	-	14.98×10^6 lb/hr (105 percent of reactor rated power)

These conditions are the most severe, because the maximum stored energy exists at these conditions. At lower power conditions the transients would be less severe.

4.3 Transients

4.3.1 The overpressure protection system must accommodate the most severe pressurization transient. Both the closure of all main steam isolation valves, and a turbine trip with bypass failure, produce severe transients. The evaluation of transient behavior with final plant configuration has shown that the isolation valve closure is slightly more severe when credit is taken only for indirect derived scrams; therefore, it is used as the overpressure protection basis event.

4.4 Scram

a. SCRAM reactivity curve - Figure 2

b. Control rod drive SCRAM motion - Figure 2

4.5 Safety/Relief Valve Transient Analysis Specifications

a. Valve groups - 5

b. Pressure setpoint - 1165-1205 psig (+ 1 percent assumed error)

4.6 Safety Valve Capacity

4.6.1 Sizing of the safety valve capacity is based on establishing an adequate margin from the peak vessel pressure to the vessel code limit (1375 psig) in response to the reference transients.

5. EVALUATION OF RESULTS

5.1 Safety Valve Capacity

5.1.1 The parametric relationship between peak vessel (bottom) pressure and safety/relief valve capacity for the MSIV transient with direct, high flux and high pressure tripped SCRAM is described in Figure 3. Design basis pressurization events in all cases result in the initiation of the direct scram signal, and the expected system response is depicted on Figure 3 for that case (trip scram). Pressures shown for flux scram will result only with multiple failure in the redundant direct scram system. Even more remotely expected are the pressures shown for pressure scram which will occur only with the multiple failures of both the redundant direct scram and the redundant flux scram systems.

5.1.2 The time response of the vessel pressure to the MSIV transient with both flux and pressure SCRAM for eighteen valves is illustrated in Figure 4. This shows that the pressure at the vessel bottom exceeds 1250 psig for less than 10 seconds which is not long enough to transfer any appreciable amount of heat into the vessel metal (which was at a temperature well below 550°F at the start of the transient).

5.1.3 From the analytical models described in Paragraph 3 together with engineering studies, it has been determined that the safety/relief valve reclosing pressures as specified in Paragraph 6.3.1.1 are acceptable.

6. SAFETY/RELIEF VALVE CHARACTERISTICS

6.1 Schematic Arrangement. The schematic arrangement of the safety/relief valves are shown in Figures 5 and 6.

6.2 Pressure Drop in Inlet and Discharge

6.2.1 Pressure drop on the piping from the reactor vessel to the valves is taken into account in calculating the maximum vessel pressures.

6.2.2 Pressure drop in the discharge piping to the suppression pool is limited by proper discharge line sizing to prevent back pressure on each safety/relief valve from exceeding 40 percent of the valve inlet pressure, thus assuring choked flow in the valve orifice, and no reduction of valve capacity due to the discharge piping. Each safety/relief valve has its own separate discharge line.

6.3 Safety/Relief Valve Description

6.3.1 These valves were manufactured by Crosby Valve and Gage Company to ASME Section III, 1971. They comply with ASME III, Paragraph NB-7640 as safety valves with auxiliary actuating devices.

6.3.1.1 Quantities, set points and associated capacities are as follows:

Quantity	Opening Set Point psig	Reclosing Pressure psig	ASME Rated Capacity at 103 percent of Set Pressure, lb/hr minimum
2	1148	1022	863,900
4	1175	1046	883,950
4	1185	1055	891,380
4	1195	1064	898,800
4	1205	1072	906,250

7. CONCLUSION

7.1 Safety requirements have long demanded very high reliability in the reactor SCRAM functions. Recognition of this reliability as being completely adequate justification for these functions to contribute to vessel pressure protection is reflected in the Section III Code provisions. Actual General Electric design practice very conservatively applies the code provisions; this results in margins even beyond those necessary to satisfy code limits, which further enhances the reliability of vessel pressure protection.

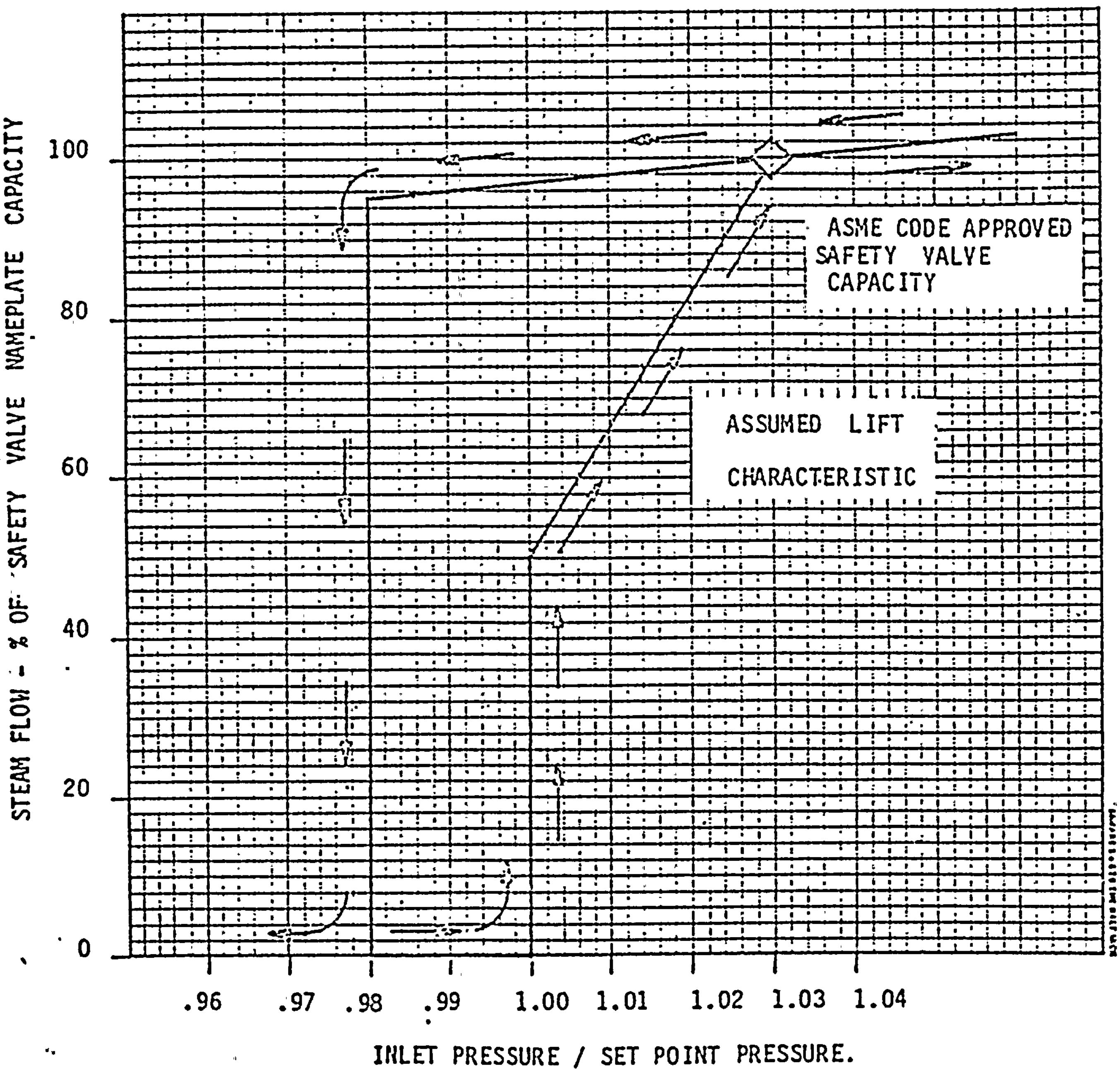

FIGURE 1 TYPICAL S/R VALVE CAPACITY CHARACTERISTIC

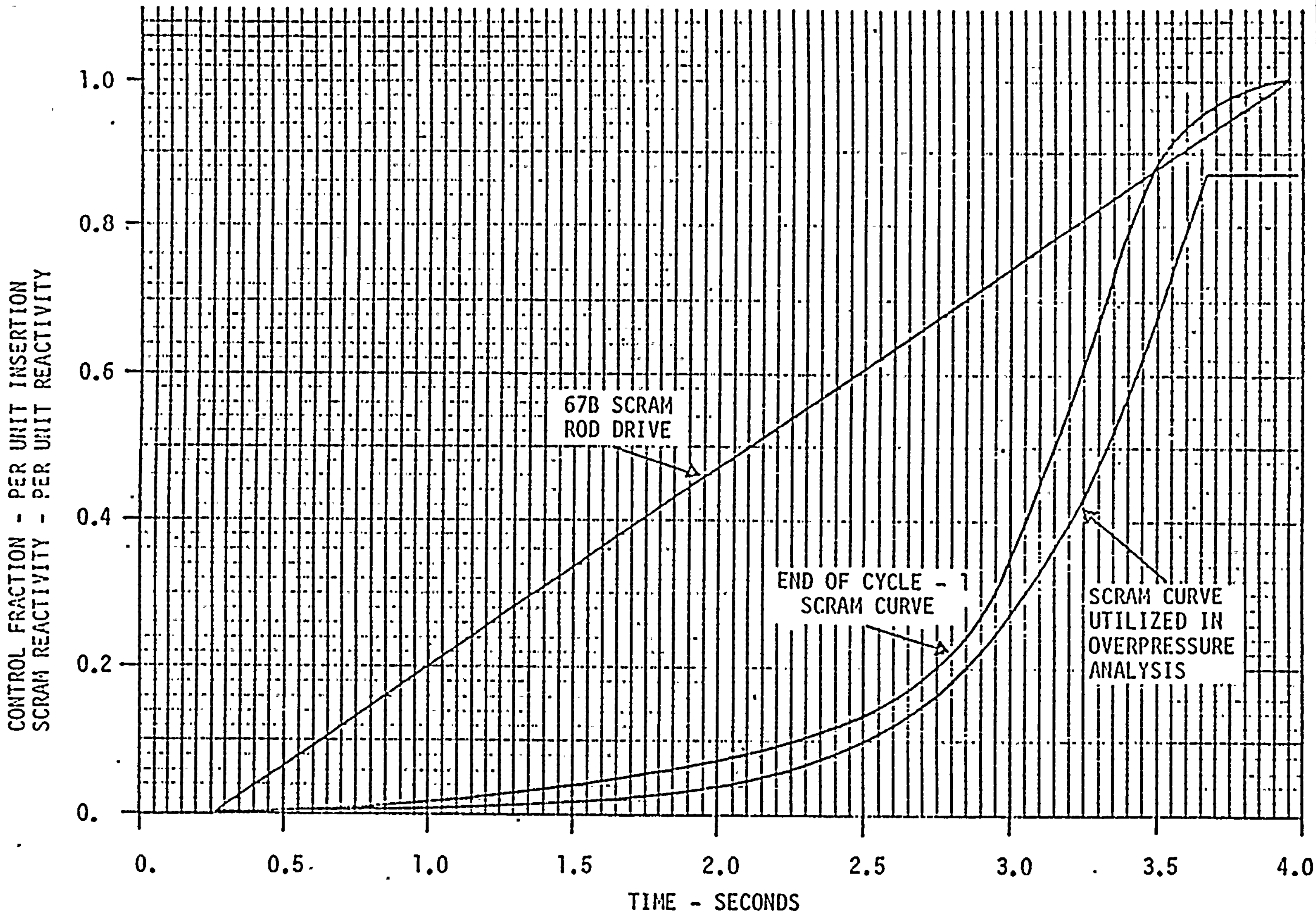

FIGURE 2 SCRAM REACTIVITY CURVE AND SCRAM ROD DRIVE VERSUS TIME

GENERAL ELECTRIC
NUCLEAR ENERGY DIVISION

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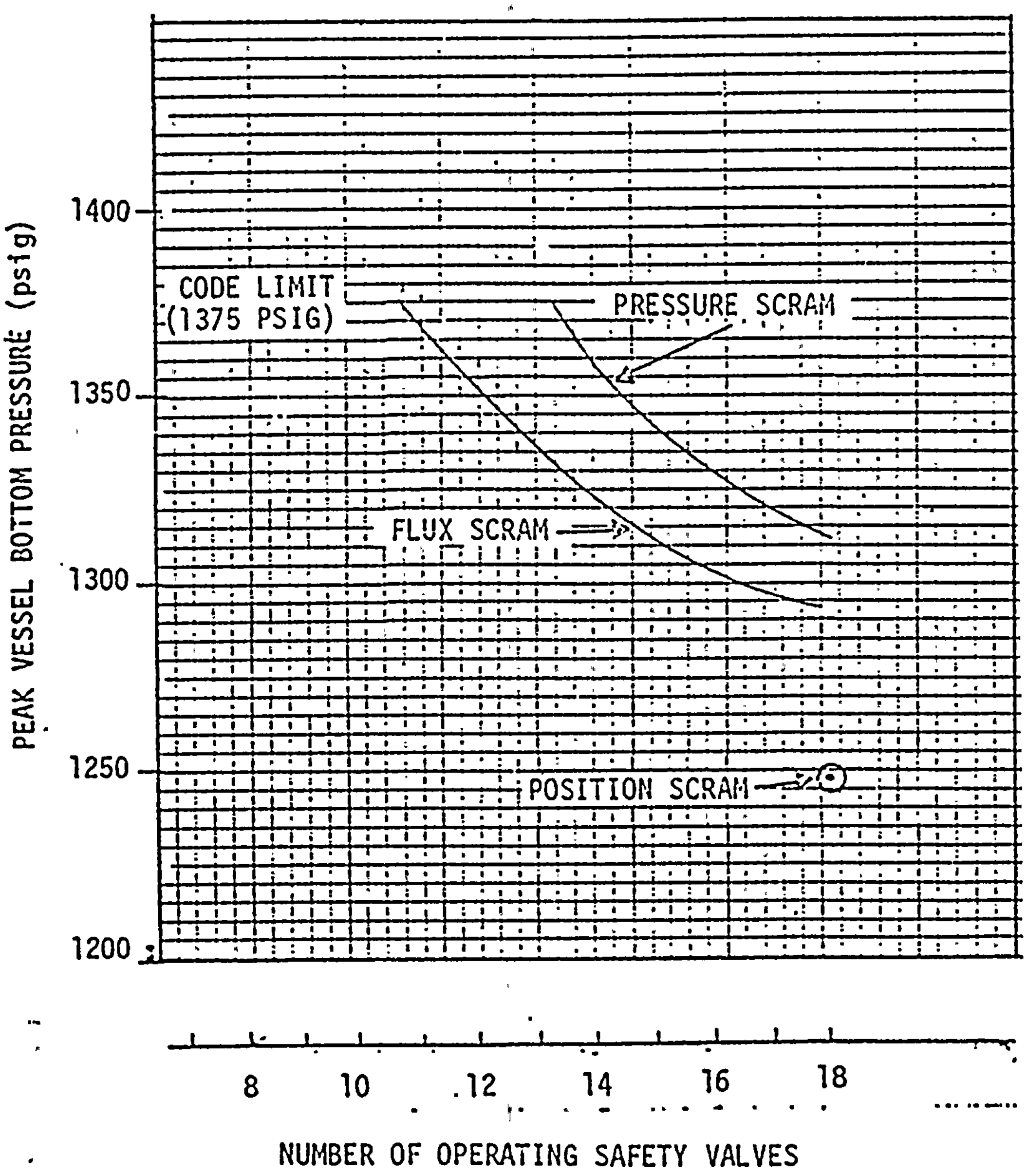

FIGURE 3 PEAK VESSEL PRESSURE VERSUS SAFETY VALVE CAPACITY

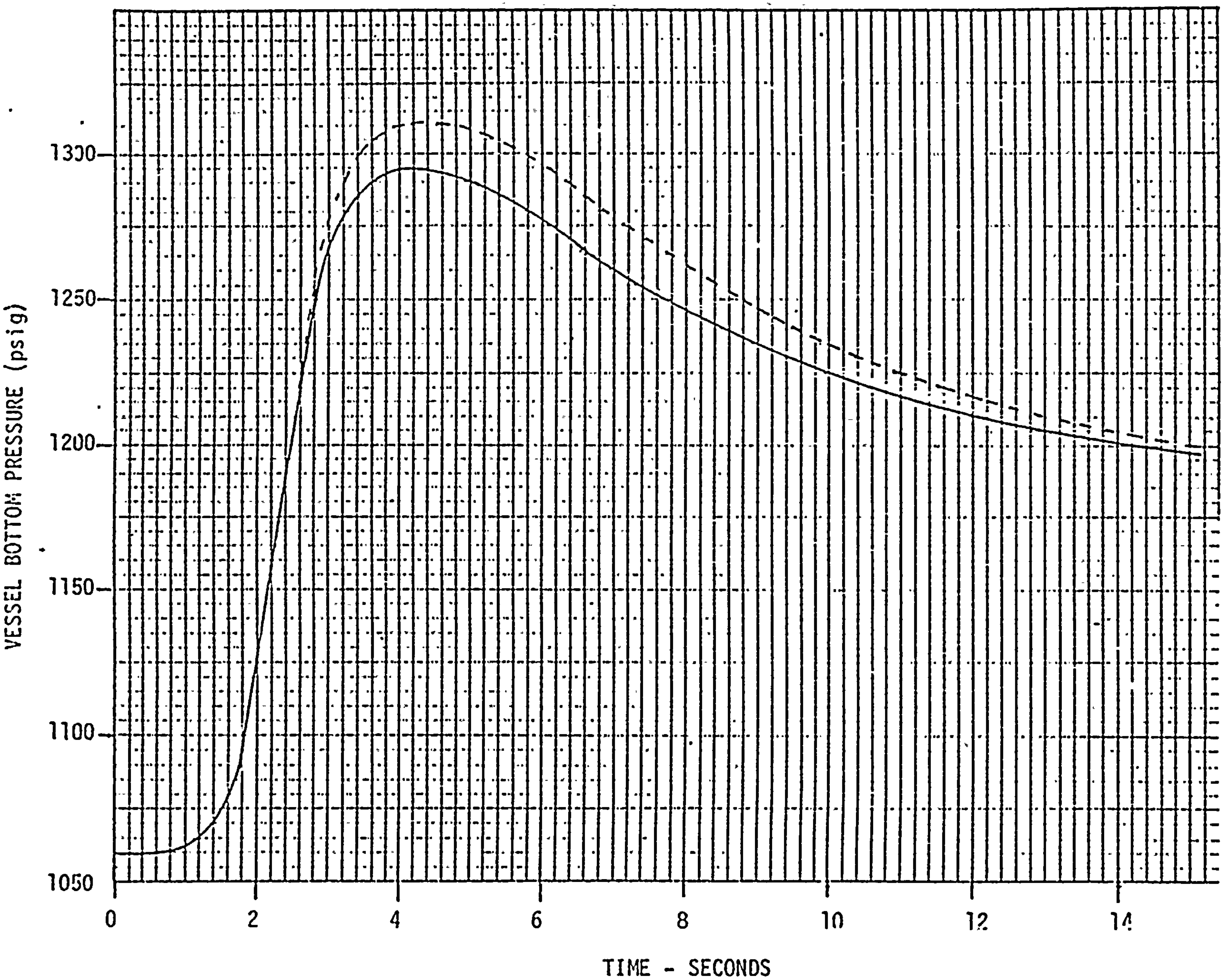

FIGURE 4 TIME RESPONSE OF VESSEL PRESSURE FOR MSIV CLOSURE TRANSIENTS

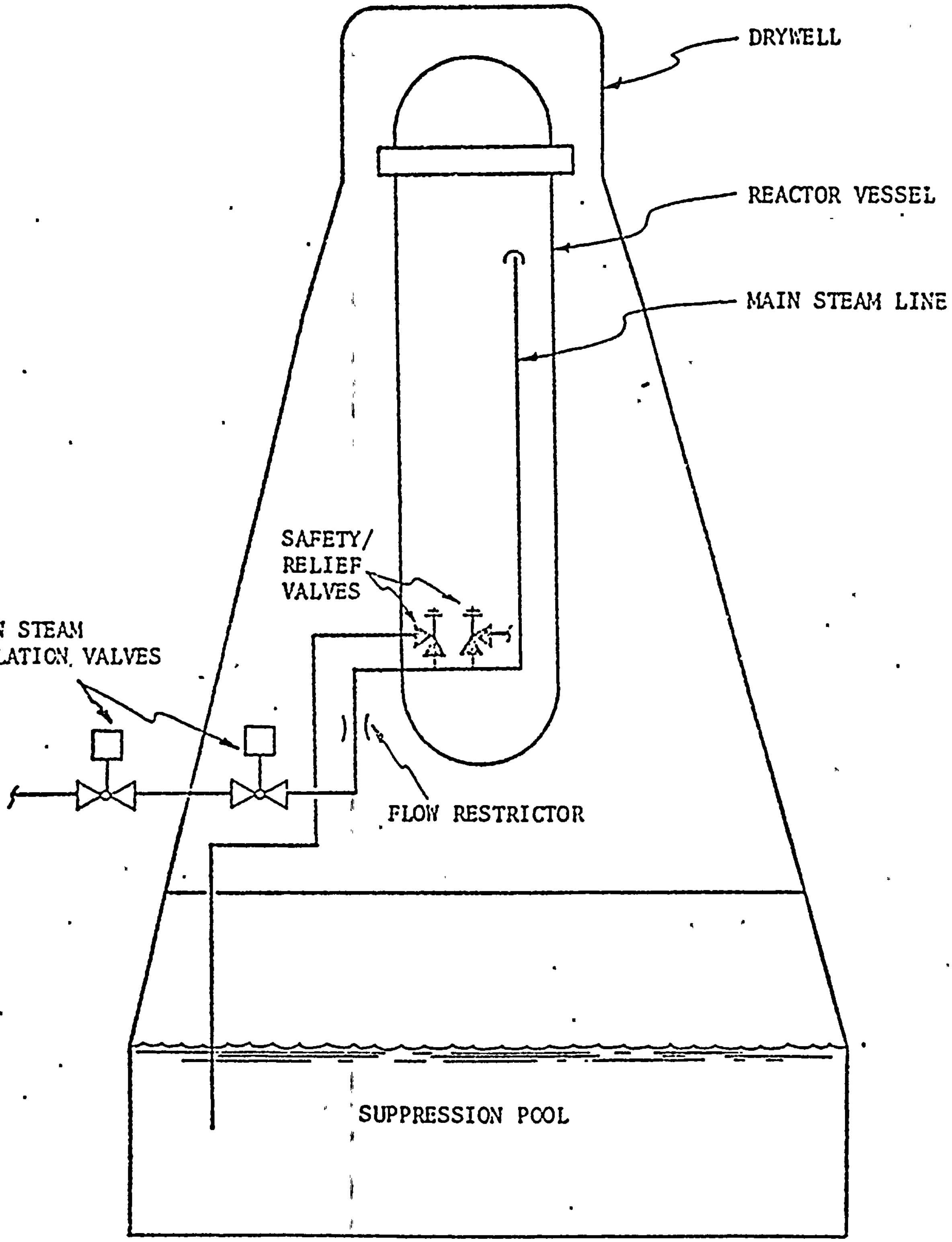

FIGURE 5 SCHEMATIC ELEVATION

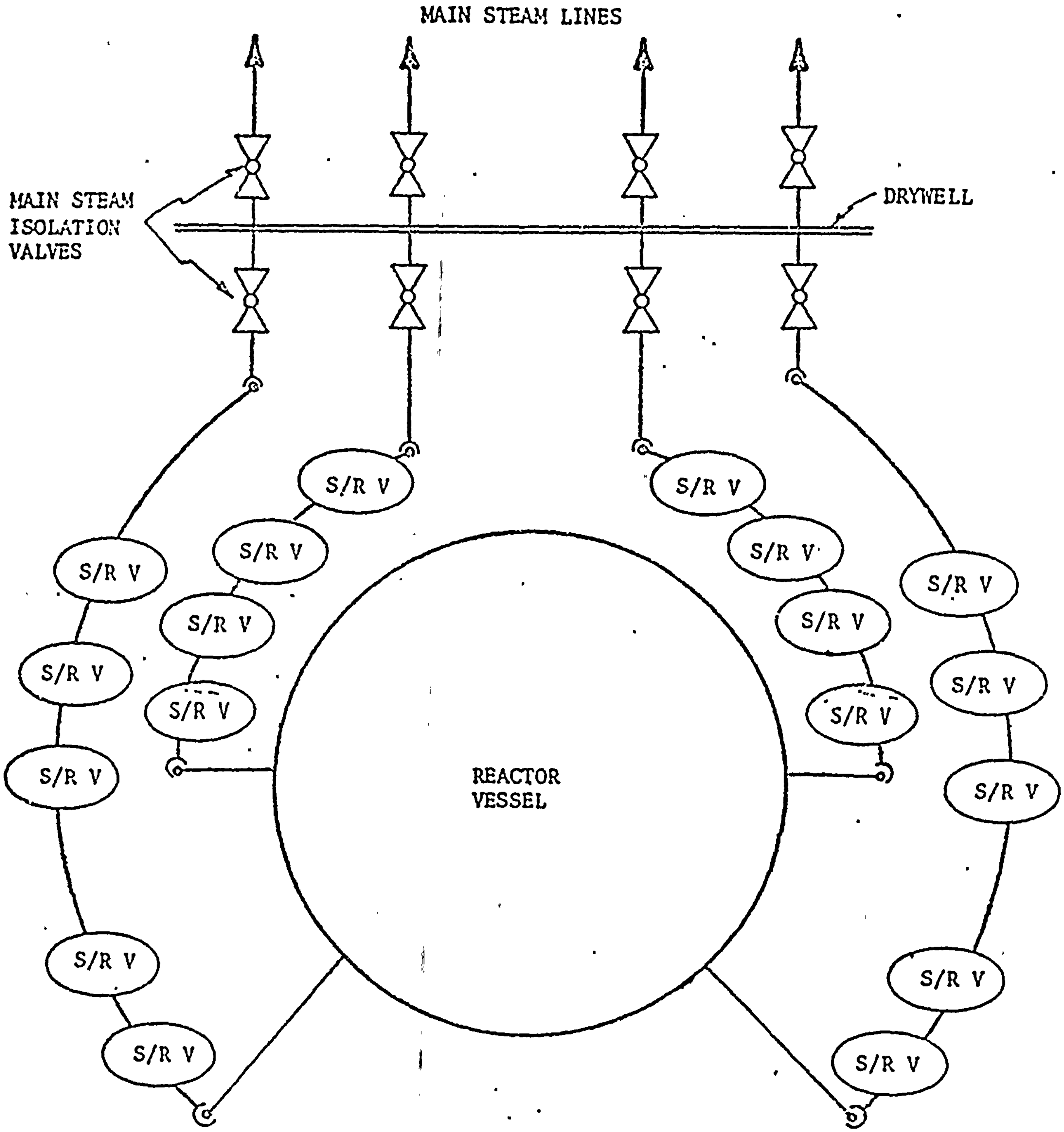

FIGURE 6 SCHEMATIC PLAN

Item #36

EQUIPMENT	SOURCE	OC PROTECTION PRIMARY	BACKUP/LOCATION	PENET
RRC-V-67A	MC-7C	25 AF	MC-7C 50 AF	X-104
RCC-V-71A	MC-7C	1.25 AF	MC-7C 25 AF	
RCC-V-72A	MC-7C	1.25 AF	MC-7C 25 AF	
RCC-V-17A	MC-7C	1.25 AF	MC-7C 25 AF	
CRA-FN-1C-2	MC-8B	110 AF	SL-81 1000A ASST	X-104B
RRC-V-67B	MC-8C	25 AF	SL-81 1000A ASST	
RRC-V-23B	MC-8C	12 AF	SL-81 1000A ASST	
RWCU-V-102	MC-8C	5 AF	SL-81 1000A ASST	
RWCU-V-106	MC-8C	3 AF	SL-81 1000A ASST	
RRC-V-23A	MC-8C	1 AF	SL-81 1000A ASST	
RWCU-V-101	MC-8C	3 AF	SL-81 1000A ASST	
RWCU-V-100	MC-8C	3 AF	SL-81 1000A ASST	
RCC-V-17B	MC-8C	1.25 AF	SL-81 1000A ASST	
RCC-V-71C	MC-8C	1.25 AF	SL-81 1000A ASST	
RCC-V-71B	MC-8C	1.25 AF	SL-81 1000A ASST	
CRA-FN-1A-2	MC-7B	110 AF	MC-7B 150 AF	X-104
MT-HOI-18	MC-3D-A	50 AF	MC-3D 200 ACB	
CRA-FN-1A-1	MC-7B	70 AF	MC-7B 90 AF	
CRA-FN-2A-2	MC-7B	50 AF	MC-7B 90 AF	
CRA-FN-2A-1	MC-7B	110 AF	MC-7B 110 AF	
CRA-FN-5A	MC-7B	25 AF	SL-71 800A ASST	
CRA-FN-4A	MC-7B	15 AF	SL-71 800A ASST	
CRA-FN-5C	MC-7B	25 AF	SL-71 800A ASST	
CRA-FN-3A	MC-7B	25 AF	SL-71 800A ASST	
MT-HOI-19C	MC-3D-A	10 AF	MC-3D 200 ACB	
* CRA-AD-1A-1	TB-R416	5 AF	MC-7B 20 AF	
* CRA-AD-1A-2	TB-R416	5 AF	MC-7B 20 AF	
CRA-AD-2A (MOTOR)	MC-7B (LATER)		SL-71 800A ASST	
CRA-AD-2A (VALVE)	MC-7B (LATER)		SL-71 800A ASST	
CRA-FN-1B-2	MC-8B	110 AF	MC-8B 150 AF	X-104
CRA-FN-1B-1	MC-8B	70 AF	MC-8B 90 AF	
CRA-FN-1C-1	MC-8B	70 AF	MC-8B 90 AF	
CRA-FN-2B-1	MC-8B	110 AF	SL-81 1000A ASST	
CRA-FN-2B-2	MC-8B	30 AF	SL-81 1000A ASST	
MS-V-16	MCC-8B-A	1.25 AF	MC-8B 125 ACB	
RWCU-V-1	MC-8B-A	5.6 AF	MC-8B 125 ACB	
RHR-V-9	MC-8B-A	15 AF	MC-8B 125 ACB	
RCIC-V-63	MC-8B-A	15 AF	MC-8B 125 ACB	
RCC-V-40	MC-8B-A	3.2 AF	MC-8B 125 ACB	
RHR-V-123B	MC-8B-A	1.125 AF	MC-8B 125 ACB	

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OC PROTECTION

EQUIPMENT	SOURCE	PRIMARY	BACKUP		PENET
RCIC-V-76	MC-8B-A	.4AF	MC-8B	125 ACB	X-104D
CRA-FN-5B	MC-8B	25AF	SL-81	1000 A ASST	
CRA-FN-5D	MC-8B	25AF	SL-81	1000 A ASST	
CRA-FN-3B	MC-8B	25AF	SL-81	1000 A ASST	
CRA-FN-3C	MC-8B	25AF	SL-81	1000 A ASST	
CRA-FN-4B	MC-8B	15AF	SL-81	1000 A ASST	
RCC-V-72B	MC-8C	1.25AF	SL-81	1000 A ASST	
MS-V-1	MC-8C-B	4 AF	MC-8C	200 ACB	
MS-V-2	MC-8C-B	4AF	MC-8C	200 ACB	
MS-V-5	MC-8C-B	4AF	MC-8C	200 ACB	
(LATER) DAMPERS	TB-R417	5AF	MC-8B	20AF	
(LATER) DAMPERS	TB-R417	5AF	MC-8B	20AF	
(LATER) DAMPERS	TB-R417	5AF	MC-8B	20AF	
(LATER) DAMPERS	TB-R417	5AF	MC-8B	20AF	
* MOTOR CRA-AD-2B	MC-8B	LATER	SL-81	1000 A ASST	
* VALVE CRA-AD-2B	MC-8B	LAER	SL-81	1000 A ASST	
RHR-V-123A	MC-8B-A	2.25AF	MC-8B	125 ACB	
RRC-P-1A	SH-5	RRA	(LATER)		X-103A
		LF2A	(LATER)		
RRC-P-1B	SH-6	RRB	(LATER)		X-103B
		LF2B	(LATER)		

* POSSIBLE DELETIONS PER PCN 7166

It is expected that the ISEG may interface with the quality assurance (QA) organization, but perferably should not be an integral part of the QA organization.

The functions of the ISEG require daily contact with the operating personnel and continued access to plant facilities and records. The ISEG review functions can, therefore, best be carried out by a group physically located onsite. However, for utilities with mulitple sites, it may be possible to perform portions of the independent safety assessment function in a centralized location for all the utility's plants. In such cases, an onsite group still is required, but it may be slightly smaller than would be the case if it were performing the entire independent safety assessment function. Such cases will be reviewed on a case-by-case basis.

At this time, the requirement for establishing an ISEG is being applied only to applicants for operating licenses in accordance with Acton Plan Item I.B.1.2. The staff intends to review this activity in about a year to determine its effectiveness and to see whether changes are required. Applicability to operating plants will be considered in implementing long-term improvements in organization and management for operating plants (Action Plan Item I.B.1.1).

WNP-2 Position

The Supply System has established a Nuclear Safety Assurance group for WNP-2 within the Licensing and Assurance Directorate as shown in Figure I.B.1.2-1. The onsite Nuclear Safety Assurance group (comprised of a minimum of one supervisor and two engineers) is supplemented by offsite technical expertise from within the Licensing and Assurance Directorate as required with a minimum of two qualified engineers available to support the WNP-2 assurance group. The WNP-2 Nuclear Safety Assurance group is independent of the line management responsible for power production and chartered with ensuring and improving operational nuclear safety of the WNP-2 plant.

The functions of the WNP-2 Nuclear Safety Assurance group include the following:

a. Evaluation of procedures important to safe operation of WNP-2 for technical adequacy and clarity.

[illegible]

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b. Evaluation of plant operations from a safety perspective.

c. Evaluation of the operating experience of WNP-2 to provide recommendations on safety-related concerns. In this regard operating experience of other plants of similar design is assessed for applicability to WNP-2.

d. Overall assessment of WNP-2 plant performance regarding conformance to safety requirements.

e. Other matters relating to safe operation of WNP-2 that independent review deems appropriate for consideration.

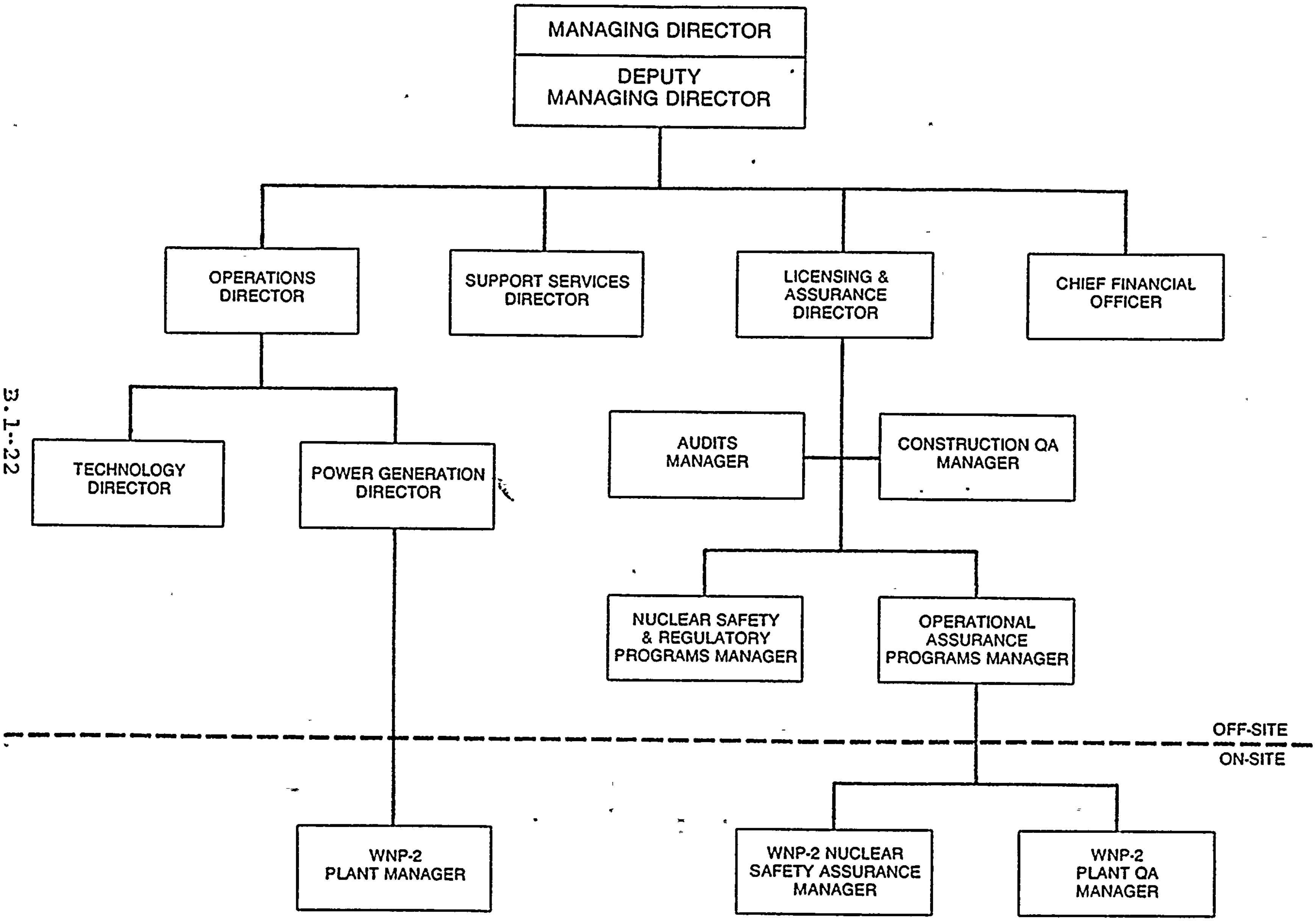
The qualification and training requirements for the Nuclear Safety Assurance Manager are comparable to that described in Section 4.2 of ANS 3.1, Draft Revision, dated March 13, 1981. Other qualifications and training requirements meet ANS 3.1, Draft Revision, dated March 13, 1981, Section 4.2 or 4.4 or equivalent as described in Section 4.1.

WASHINGTON PUBLIC POWER
SUPPLY SYSTEM
NUCLEAR PROJECT NO. 2

NUCLEAR SAFETY ASSURANCE ORGANIZATION

FIGURE
I.B.1. 2-1

B.1-22



Item # 44

AMENDMENT NO. 30
June 1983

BURNS AND ROE, INC.
Headquarters Office—Oradell, N.J.

Item # 49

W.O. No. ______ Calc. No. ______ Sheet ______ Cont. on Sheet ______

Title ______

PURPOSE

To document and, when necessary, establish trip setpoints for those instruments in this system which perform a tripping action.

REFERENCES

Specific references are identified on the individual data sheet for each instrument.

DESIGN REQUIREMENTS

1) US NRC Regulatory Guide 1.105, Revision 1, November 1976, Instrument Setpoints.

2) Conform with the instrument purpose with regard to the system design.

3) Conform with vendor recommendations when provided.

ASSUMPTIONS

None

PROCEDURES

1) Total loop inaccuracies can be calculated by either using the sum of the loop component inaccuracies or the square root of the sum of the squares of the loop component inaccuracies. Where SRSS is used, it shall be so noted.

2) If required, additional system performance calculations may be performed in cases where system and instrumentation incompatabilities occur. In these cases, the performance calculations will become an integral part of the setpoint calculation and will be attached to the data sheet(s), OR if applicable, an original calculation may be revised and so referenced.

3) The setpoint data sheet (Exhibit I), the instruction for the setpoint data sheet (Exhibit II) and the setpoint data sheet definitions (Exhibit III) together make up the procedure to be used for documentating and/or establishing trip setpoint calculations.

COMPUTER INPUT/OUTPUT

None

CONCLUSIONS

See attached data sheets

0	Original Issue				
Rev. No.	Description of Revision	Type (Prelim Design, Final Design, Study)	Originator Signature/Date	Checker Signature/Date	Approver Signature/Date

EXHIBIT I

Item # 49

BURNS AND ROE, INC.
Headquarters Office—Oradell, N.J.

W.O. No. ____ Date ____ Book No. ____ Page No. ____
Drawing No. ____ Calc. No. ____ Sheet ____ Cont. on Sheet ____
By ____ Checked ____ Approved ____
Title SETPOINT DATA SHEET

1. INSTRUMENT NO. ____ SWITCH NO. ____ CIRCUIT NO. ____ NO/NC CONTACT

   MFG/MODEL ____ INST. RANGE ____

2. FUNCTION ____

   A) TECH. SPEC. OR PROTECTIVE LIMIT (YES/NO) IF YES, GO TO STEP 3.

   B) PROCESS FUNCTION, ALARM OR INDICATION (YES/NO) IF YES, COMPLETE STEPS 7, 12 &13.

   C) OTHER ____ (YES/NO) IF YES, JUSTIFY METHODOLOGY IN REMARKS (STEP 13).

3. ALLOWABLE VALUE ____ (TECH SPEC ONLY) TRIP SETPOINT ____

   REFERENCE ____ SEE INSTRUCTION. IF NO ACCEPTABLE REFERENCE COMPLETE STEPS 4-13. OTHERWISE GO TO STEP 13.

4. PROCESS READOUT ACCURACY

COMPONENT	MFG/MODEL	ACCURACY (%)		CALIBRATED RANGE		INSTRUMENT ACCURACY	REFERENCE
			X		=		
			X		=		
			X		=		
TOTAL LOOP ACCURACY							

5. INSTRUMENT DRIFT ALLOWANCE ____ ____

6. RESET DEADBAND ____ ____

7. NORMAL OPERATING POINT/RANGE ____ ____

8. ○ MAXIMUM/○ MINIMUM OPERATING POINT (VALUE TOWARD LIMITING CONDITION) ____ ____

9. ANALYTICAL LIMIT (AL) ____ ____

10. ALLOWABLE VALUE (AV) (STEP 9±STEP 4) ____ SEE INSTRUCTION

11. TRIP SETPOINT LIMIT (STEP 10±STEP 5) ____ SEE INSTRUCTION

12. TRIP SETPOINT REFERENCE ____ ____ SEE INSTRUCTION

13. CONCLUSIONS: TRIP SETPOINT ____ (PROCESS UNITS) ON: INCREASE/DECREASE

    TRIP SETPOINT ____ (INSTRUMENT UNITS)

    ALLOWABLE VALUE ____

    REMARKS:

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EXHIBIT II

BURNS AND ROE, INC.

Headquarters Office—Oradell, N.J.

Item # 49

W.O. No. ______ Date ______ Book No. ______ Page No. ______

Drawing No. ______ Calc. No. ______ Sheet ______ Cont. on Sheet ______

By ______ Checked ______ Approved ______

Title INSTRUCTION

STEP 1. Complete all required information. If not applicable, mark NA.

STEP 2. Explain the function of the switch. Complete one of the following.

A) Identify if a Technical Specification applies or if the switch fuction is a protective limit. A protective limit is one which results in equipment damage or an unsafe condition if exceeded. A tech. spec. or protective limit can also be in the form of a process function, alarm or indication, however, this step must be complied with.

B) Identify setpoints which are to control process functions, alarm that a condition exists or provide an indication. Instruments in this catagory are operator aids and are not tech. spec. or protective limit.

C) Identify and explain any additional catagories of instruments and select appropriate calculation method.

STEP 3. Enter allowable values (tech. spec. only) and trip setpoints as defined in definitions. Reference their respective engineering documents (e.g. drawing, calculation, vendor information). If this data is unavailable, STEPS 4-13 must be used to derive these values.

STEPS 4-9. Enter the required information and references. See Definitions.

STEP 10. Calculate the Allowable Value (AV). If a minimum limit (STEP 8) then, AV=STEP 9+STEP 4, if a maximum limit, AV=STEP 9-STEP 4.

STEP 11. Calculate the Trip Setpoint Limit. This value as calculated is as close as the setpoint can be to (AL). This value can be the actual setpoint or it may be established using the guidelines in STEP 12.

STEP 12. Setpoints must be established to perform the required function yet minimize inadvertent tripping due to operational transients. The following guidelines should apply or an improper instrument may be applied.

STEP 8 (MAX) + STEP 4 + STEP 6 < TRIP SETPOINT, OR

STEP 8 (MIN) + STEP 4 - STEP 6 > TRIP SETPOINT

If STEP 2B, or possibly 2C, is used, reference the source of or method used to determine the setpoint. Continue in Remarks section if necessary.

STEP 13. Enter the Trip Setpoint, both in process and instrument units, the Allowable Value (Tech. Spec. only), and note any Remarks which may be helpful in understanding how this calculation was completed.

PH 18 [illegible]

EXHIBIT III

BURNS AND ROE, INC.

Headquarters Office—Oradell, N.J.

KM #49

W.O. No. ______ Date ______ Book No. ______ Page No. ______

Drawing No. ______ Calc. No. ______ Sheet ______ Cont. on Sheet ______

By ______ Checked ______ Approved ______

Title DEFINITIONS

1) Accuracy: Manufacturers rated accuracy in percent full scale.

2) Accuracy, Total Loop: The sum of the instrument accuracies or, if desired, the square root of the sum of the squares of the instrument accuracies.

3) Analytical Limit (A.L.): The value of the sensed process variable prior to which a desired action is to be initiated to prevent the process variable from reaching the associated design/safety limit.

4) Allowable Value: Analytical Limit minus Total Loop Accuracy.

5) Instrument Drift - Estimated change in instrument setpoint between calibrations.

6) Maximum Operating Point - Normal plant process operating point plus normal operating transients.

7) Minimum Operating Point - Normal plant process operating point minus normal operating transients.

8) NO/NC Contact - Normally Open/Normally Closed contact used for switch function.

9) Reset Deadband - Contact deadband between trip setpoint and contact reset (return to normal).

10) Switch No. & Circuit No. - Identifies the switch and circuit within the instrument as identified by the instrument manufacturer.

11) Trip Setpoint Limit - This value is as close to the Analytical Limit as the instrument can initially be set.

P44 #P msg/