SOUTH CAROLINA ELECTRIC & GAS COMPANY
POST OFFICE BOX 764
COLUMBIA, SOUTH CAROLINA 29218

T. C. NICHOLS, JR.
VICE PRESIDENT AND GROUP EXECUTIVE
NUCLEAR OPERATIONS

February 19, 1981

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Subject: Virgil C. Summer Nuclear Station
Docket No. 50/395
Reactor Systems Branch - Open Items

Dear Mr. Denton:

The following is a discussion of open items from the Reactor Systems Branch regarding NUREG-0737 and other miscellaneous items.

1. II.B.1 - Reactor Vessel Vents

FSAR Section 5.5.15 gives the major portion of the responses to this item. The attached marked up pages (which are included in Amendment 24 to be issued on March 2, 1981) provides responses to open items discussed with Mr. Chu Liang. FSAR Section 5.5.13 provides a discussion on venting the pressurizer. The attached marked up table 1.8.1 (Amendment 24) shows reference to that section. The attached generic procedures "Reactor Vessel Head Vent Operation" from the Westinghouse Owner's Group provides guidelines to our operations group to develop plant procedures for operating the vessel vent system. Plant specific guidelines are due to us by March 31, 1981. Plant procedures will be sent to you for review within the next few months. This will be well before the requirement of January 1, 1982. Finally, the system is being installed in the plant at this time. This will meet the requirement of installation by July 1, 1982.

2. II.K.2-13 - Thermal Mechanical Report

NUREG-0737 item II.K.2-13 allows operating plants to complete their required analyses by January 1, 1982, but requires applicants for operating licenses to submit analyses six (6) months prior to issuance of the staff Savety Evaluation Report (SER) for a full power license. The staff SER was issued in February 1981; six (6) months prior to this date is September 1980. This date proceeds the date of issuance of NUREG-0737.

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II.K.2-13 Thermal Mechanical Report (con't)

SCE&G requests that the required submittal date for our analyses be shifted to January 1, 1982, the date for OL licensees, for the following reasons:

1. During the period to January 1, 1982 the Virgil C. Summer Nuclear Station reactor vessel will have undergone an insignificant amount of irradiation damage. However, some Westinghouse reactor vessels have experienced more than ten years of operation. If the NRC finds the January 1, 1982 date acceptable for these older vessels, SCE&G concludes that the NRC should find the January 1, 1982 date acceptable for the Virgil C. Summer Nuclear Station.

2. The Westinghouse Owner's Group is embarking on a program to evaluate plants on a generic basis. Plant specific analyses, if required, will be based on the results of the generic analyses. SCE&G believes that a more comprehensive analyses and the best results can be obtained for the Virgil C. Summer Nuclear Station vessel by participation in this generic program. Generic results will be presented by January 1, 1982.

3. II.K.3-30 - Revised Small Break LOCA Model

In our January 6, 1981 letter a discussion was provided on the revised small break LOCA model. Current plans by the Westinghouse Owner's Group are to have this revised model submitted to the NRC by January 1, 1982.

4. II.K.3-31 - Plant Specific Calculations to Show Compliance with 10 CFR Part 50.46

In our January 6, 1981 letter a discussion was provided on the plant specific analyses. If the results of the new model indicate that a plant specific small break LOCA analyses is required to conform with the requirements with 10 CFR 50.46, then it will be submitted by January 1, 1983.

5. II.K.3-2 - Automatic Power Operated Relief Valve Operation

Current plans by the Westinghouse Owner's Group are to have the report issued by March 15, 1981. This was discussed with Mr. Brian Sheron by the Westinghouse Owner's Group on February 17, 1981.

6. II.K.3-17 - ECCS Outages

The attached marked up FSAR Section 13.5.1.14 is provided to respond to this item. This material will be included in FSAR Amendment 24.

Also included in this letter are several minor changes in the boron dilution evaluation in Chapter 15.2. This material, which will be included in FSAR Amendment 24, is provided in order to expedite its review.

If you have any questions, please let us know.

Very truly yours,

T. C. Nichols, Jr.

T. C. Nichols, Jr.

RBC:TCN:rh

cc: V. C. Summer
G. H. Fischer
T. C. Nichols, Jr.
E. H. Crews, Jr.
O. W. Dixon, Jr.
C. A. Price
D. A. Nauman
W. A. Williams, Jr.
R. B. Clary
A. R. Koon
A. A. Smith
H. N. Cyrus
J. B. Knotts, Jr.
J. L. Skolds
B. A. Bursey
O. W. Bradham
ISEG
PRS
NPCF/Whitaker
File
R. Faas

TABLE 1.8-1

CROSS REFERENCE

TMI ACTION PLAN REQUIREMENTS TO FSAR SECTIONS

ACTION PLAN REQUIREMENT	FSAR SECTION	
I.A.1.1	13.1.2.1, 13.1.2.2.2, 13.1.3.1.26, 13.2.1.7.7	
I.A.1.2	13.1.2.2, 13.5.1.3.1	
I.A.1.3	13.1.2.3, 13.5.1.3	
I.A.2.1	13.2.1.1.6, 13.2.3.1, SCE&G letters to NRC dated 10/28/80, 10/31/80	
I.A.2.3	13.2.2, SCE&G letter to NRC dated 10/28/80	
I.A.3.1	13.2.1.1.6, 13.2.2.6.1, SCE&G letter to NRC dated 10/28/80	23
I.B.1.2	13.1.2.1, 13.5.1.3.1, SCE&G letter dated 1/2/81	
I.C.1	6.3.3.3.1, 13.5.2, SCE&G letters dated 11/14/80, 12/2/80, and Westinghouse Owners Group Letter OG-47 dated 12/15/80	
I.C.2	13.5.1.3	
I.C.3	13.5.1.3.1, Tech Spec (6.1.2)	
I.C.4	13.5.1.3	
I.C.5	13.5.1.13	
I.C.6	13.5.1.6	
I.C.7	13.5.1.3.2	
I.C.8	13.5.1.3.3, SCE&G letter to NRC dated 11/14/80	
I.D.1	1.2.3.1	
I.D.2	7.7.3	
I.G.1	14.1.4.4	
II.B.1	5.5.15, Question 211.133, 5.5.13, SCE&G letter dated 2/19/81.	24
II.B.2	12.1.2.3, App. 12A, SCE&G letters to NRC dated 8/27/80 and 11/21/80	

TABLE 1.8-1 (Continued)

CROSS REFERENCE

TMI ACTION PLAN REQUIREMENTS TO FSAR SECTIONS

ACTION PLAN REQUIREMENT	FSAR SECTION
II.B.3	9.3.2, App. 12A, Responses to Questions 321.13 and 321.16
II.B.4	13.2.1.1.6
II.D.1	5.5.13.4
II.D.3	1.2.3.1, 1.7, 5.5.10.2.2.4, 5.6, 7.7.4, SCE&G letter to NRC dated 1/13/81
II.E.1.1	SCE&G letters to NRC dated 8/15/80, 11/5/80 and 12/2/80 23
II.E.1.2	7.3.1.1.1, 7.3.2.2, 7.3.2.3, 7.5.1, 10.4.9.1, 10.4.9.2, 10.4.9.3, 10.4.9.5
II.E.3.1	8.3.1.1.2.a
II.E.4.1	6.2.5.2.1
II.E.4.2	6.2.4.3; 9.4.8.2.2, Items 8.1 and 8.m; 9.4.8.2.3, Items 8.h and 8.8
II.F.1	6.2.5.1.3, 6.2.5.2.3, 6.2.5.3.3, 6.2.5.4.3, 6.2.5.5.3, 6.2.5.5.4, Table 7.5-1, 7.7.3.1.c, 10.4.2, 11.4, 11.4.2., Figure 11.4-2, 12.2.5, Figure 12.2-2, 12.3.2.2, Response to Question 321.14, SCE&G letters to NRC dated 8/28/80 and 12/22/80
II.F.2	1.2.3.1, 5.6, 7.7.5, 7.7.6, SCE&G letters to NRC dated 12/4/80, 12/15/80, and 12/30/80
II.G.1	7.4.1.2.1, 8.3.1.1.3
II.K.1	7.3.1.1, 13.5.1.6, and Technical Specifications
II.K.2	SCE&G letters to NRC dated 1/6/81, 2/19/81. 24

TABLE 1.8-1 (Continued)

CROSS REFERENCE

TMI ACTION PLAN REQUIREMENTS TO FSAR SECTIONS

ACTION PLAN REQUIREMENT	FSAR SECTION
II.K.3	5.5.2.3; 7.2.1.1.2, Item 6; Figure 7.7-4; Technical Specifications, SCE&G letters to NRC dated 1/6/81, and 2/19/81 and Response to Question 211.123, 13.5.1.3.4, 13.5.1.14
III.A.1.1	Radiation Emergency Plan
III.A.1.2	1.2.3.1, 6.4, 7.7.3, 12.1.4.2, 12.3.2.2.4, 15.4.1.3
III.A.2	2.3.3.2, Radiation Emergency Plan
III.D.1.1	6.3.2.11.3, Response to Questions 321.12 and 321.15, Technical Specifications
III.D.3.3	6.2.5.1.3, 6.2.5.2.3, 6.2.5.3.3, 6.2.5.4.3, 6.2.5.5.4, 12.1.4.2, 12,3.2.2.4, Response to Question 331.43
III.D.3.4	2.2.1, 2.2.2, 2.2.3, 6.4, 15.4, SCE&G letters to NRC dated 11/25/80, 12/15/80

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5.5.15 REACTOR VESSEL HEAD VENT SYSTEM

5.5.15.1. Design Basis

The basic function of the Reactor Vessel Head Vent System (RVHVS) is to remove noncondensable gases or steam from the reactor vessel head. This system is designed to mitigate a possible condition of inadequate core cooling or impaired material circulation resulting from the accumulation of noncondensable gases in the RCS. The design of the RVHVS is in accordance with the requirements of NUREG-0578 and subsequent definitions and clarifications (References 2 and 3).

5.5.15.2 Design Description and Evaluation

5.5.15.2.1 General Description

The RVHVS is designed to remove noncondensable gases or steam from the reactor coolant system via remote manual operations from the control room. The system discharges to the pressurizer relief tank. The RVHVS is designed to vent a volume of hydrogen, non-condensible gases, etc., at system design pressure and temperature approximately equivalent to one-half of the reactor coolant system volume in one hour.

A H_2 burn from a 100% zircalloy/H_2O reaction has been addressed and it has been determined that containment integrity would not be breached. Therefore, contained venting outside containment is not considered necessary.

The flow diagram of the RVHVS is shown in Figure 5.5-13. The RVHVS consists of two parallel flow paths with redundant isolation valves in each flow path. The venting operation uses only one of the flow paths at any time. The physical layout of the RCS hot leg piping is such that its entire volume can be vented via the RVHV system.

The equipment design parameters are listed in Table 5.5-16. As indicated above, normally the venting from the RVHV is contained by the Pressurizer Relief Tank. However, venting to containment could occur if the rupture disc ruptures. In that case the location of the PRT is such that excellent gas communication exists within the secondary shield are that any gas that escapes from the PRT will be readily mixed with the containment atomosphere with additional mixing being promoted by the Reactor Building Ventilation System. The active portion of the system consists of four two-inch motor operated isolation valves connected to the reactor vessel head vent pipe. The isolation valves in series in each flow path are powered by opposite vital power supplies. The isolation valves are fail as is, active valves. One normally closed isolation valve and one normally open valve are located in each flow path. Leakage past the vent valves during normal plant operation is detected by the acoustic leak monitoring system which is described in Section 7.6.9. All of the isolation valves are qualified to IEEE-323-1974, 344-1978, and 382-1972 and to the requirements of Regulatory Guide 1.48 as described in Appendix 3A.

If one single active failure prevents a venting operation through one flow path, the redundant path is available for venting. Similarly, the two isolation valves in each flow path provide a single failure method of isolating the venting system. With two valves in series, the failure of any one valve or power supply will not inadvertently open a vent path. Thus, the combination of safety grade train assignments and valve failure modes will not prevent vessel head venting nor venting isolation with any single active failure.

The RVHVS has two normally de-energized valves in series in each flow path. Power lockout capability to all four isolation valves is provided by administrative control at the motor control load center. This arrangement eliminates the possibility of a spuriously opened flow path.

The system is operated from the control room. The position indication from each valve is monitored in the control room by status lights.

The RVHVS is connected to the head vent pipe as shown on Figure 5.5-13. The system is orificed to limit the blowdown from a break downstream of either of the orifices to within the capacity of one of the centrifugal charging pumps.

A break of the RVHVS line upstream of the orifices would result in a small LOCA of not greater than one inch diameter. Such a break is similar to those analyzed in WCAP-9600 (Reference 4). Since a break in the head vent line would behave similarly to the hot leg break case presented in WCAP-9600, the results presented therein are applicable to a RVHVS line break. This postulated vent line break, therefore, results in no calculated core uncovery.

All piping and equipment from the vessel head vent up to and including the second isolation valve in each flow path are designed and fabricated in accordance with ASME, Section III, Class 1 requirements. The remainder of the piping and equipment is non-nuclear safety, but is seismically supported up to the 12" pressurizer relief line.

The system provides for venting the reactor vessel head by using only safety grade equipment. The RVHVS satisfies applicable requirements and industry standards including ASME Code classification, safety classification single-failure criteria, and environmental qualification.

5.5.15.2.2 Supports

The vent system piping is supported to ensure that the resulting loads and stresses on the piping and on the vent connection to the vessel head are acceptable.

The support design for attaching the head vent system piping to the reactor vessel head lifting leg is shown in Figure 5.5-14. The support is a two-part clamp configuration, called a double bolt riser clamp. The clamp and associated bolts, nuts, spacers, and washers are made of stainless steel. A gap exists between the one inch head vent pipe and the support clamp to allow for thermal expansion in the vertical direction.

The support design for attaching the head vent system piping to the CRDM Seismic Support Platform is shown in Figure 5.5-15. This support is a two-part clamp configuration, called a double bolt clamp bracket. This clamp support is used to rigidly support the piping in the radial direction. The clamp and associated bolts, nuts, spacers, and washers are made of stainless steel, with high strength hold down bolts threaded into the deck of the CRDM Seismic Support Platform. A gap exists between the one inch head vent pipe and the support clamp to allow for thermal expansion in the axial direction.

All supports and support structures comply with the requirements of the AISC Code, Part II.

5.5.15.3 Analytical Considerations

The analysis of the reactor vessel head vent piping is based on the following plant operation conditions defined in the ASME Code Section III:

1. Normal Condition:

   Pressure deadweight, and thermal expansion analysis of the vent pipe during a) normal reactor operation with the two inboard vent isolation valves closed and b) post-refueling venting.

2. Upset Condition:

   Loads generated by the Operating Basis Earthquake (OBE) response spectra.

3. Faulted Condition:

   Loads generated by the Safe Shutdown Earthquake (SSE) and by valve thrust during venting. In accordance with ASME III, faulted conditions are not included in fatique evaluation.

The Class I piping used for the reactor vessel head vent is one inch schedule 160 and, therefore, in accordance with ASME Section III, is analyzed following the procedures of NC-3600 for Class II piping.

For all plant operating conditions listed above, the piping stresses are shown to meet the requirements of equations (8), (9), (10) or (11) of ASME III, Section NC-3660, with a design temperature of 650°F and a design pressure of 2485 psig.

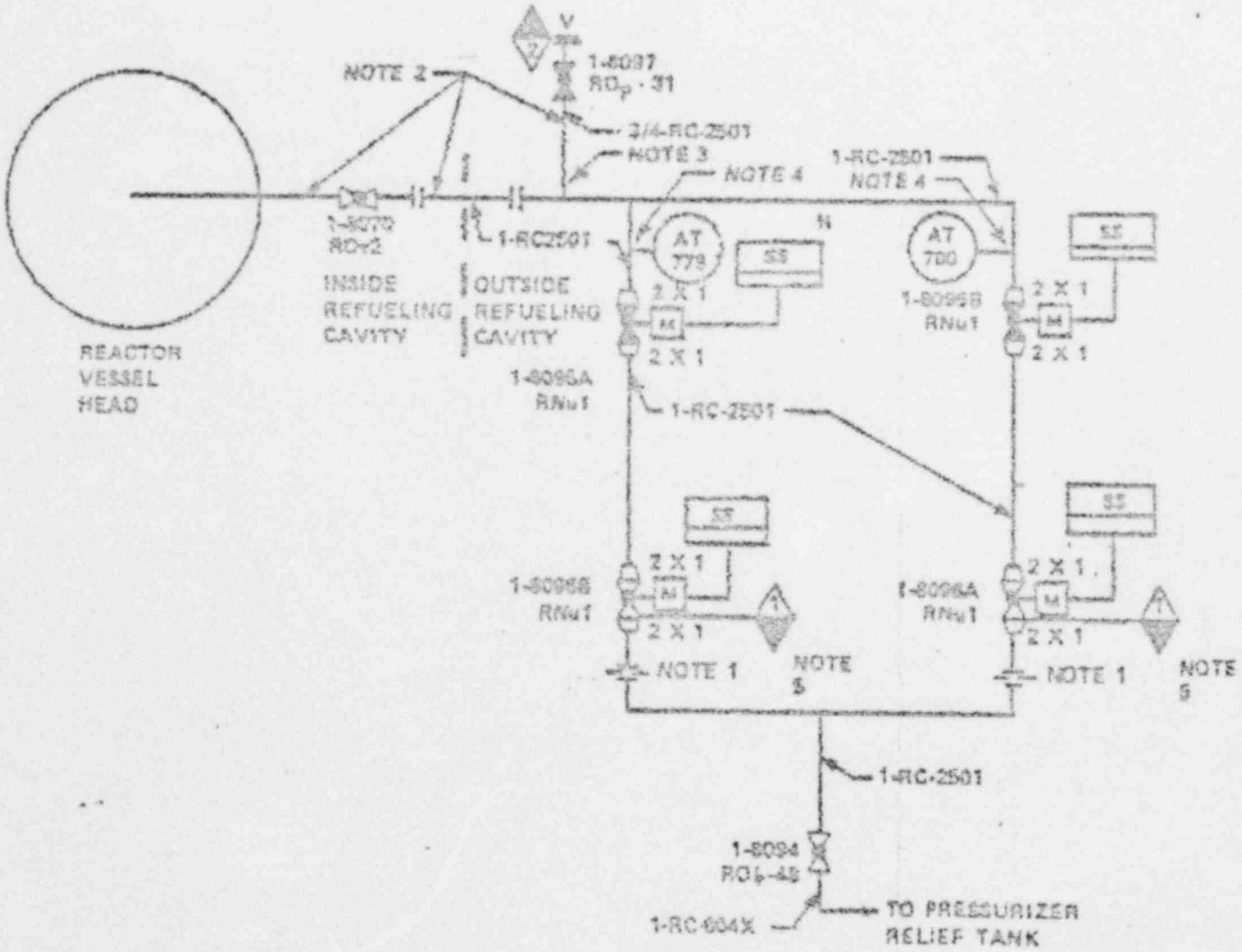
5.5.16 REFERENCES

1. "Reactor Coolant Pump Integrity in LOCA", WCAP-8163, September, 1973.

2. Letter from D. B. Vassallo (NRC) to all Applicants for an Operating License, "Followup Actions Resulting From the NRC Staff Reviews Regarding the Three Mile Island Unit 2 Accident", and Enclosure 4: Installation of Remotely Operated High Point Vents in the Reactor Coolant System, September 27, 1979.

3. Letter from D. B. Vassallo (NRC) to all Applicants for an Operating License, "Discussion of Lessons Learned Short Term Requirements," Enclosure 1, pp. 44-49, Reactor Coolant System Venting, November 9, 1979.

4. "Report on Small Break Accidents for Westinghouse NSSS System," WCAP-9600, June, 1979, (specifically Case F, Section 3.2).



NOTES:

1. 3/8 INCH FLOW RESTRICTING ORIFICE.
2. NORMAL VENT LINE.
3. 3/8 INCH FLOW RESTRICTION
4. STRAP-ON ACOUSTIC MONITORING SENSOR
5. PIPE FABRICATED TO CODE CLASS 2 BUT NOT STAMPED.

Figure 5.5-13. Schematic Flow Diagram of the Reactor Vessel Head Vent System

REACTOR VESSEL

HEAD VENT OPERATION

REVISION 0

FEBRUARY, 1981

REACTOR VESSEL HEAD VENT OPERATION

A. PURPOSE

The objective of these instructions is to specify required operator actions and precautions necessary to remove gases from the reactor vessel head by operation of the Reactor Vessel Head Vent.

CAUTION: This venting guideline should not be used as the primary means to mitigate an Inadequate Core Cooling event. Refer to Inadequate Core Cooling Guidelines for appropriate operator actions and precautions.

CAUTION: This venting guideline assumes that the reactor containment conditions are near normal conditions and that any venting operation is performed prior to throttling safety injection flow during a POST-LOCA cooldown and depressurization operation.

B. SYMPTOMS

For plants with a RV level indication

1. Reactor vessel level is less than (insert plant specific value which includes an allowance for normal channel accuracy) percent of span.

For plants with/without a RV level indication

2. Abnormal reactor coolant system conditions such as large variations in pressurizer level during normal charging of spraying operations have occurred.

3. If available, reactor vessel head temperatures equal to or greater than saturation temperature.

4. Plant events have occurred (such as accumulator tank discharge, rapid RCS cooldown, or core uncovery events) that may result in the presence of a gaseous void in the vessel head.

C. IMMEDIATE ACTIONS

None

D. SUBSEQUENT ACTIONS

CAUTION: Do not trip any running or start any non-operating reactor coolant pumps during the performance of the following actions.

NOTE: If the safety injection system is in operation, then the actions of steps marked by an asterisk will not be applicable.

1. Terminate any changes to the reactor coolant system that may be in progress and bring the RCS to as close to a steady-state condition as possible.

*2. Attempt to recombine any condensible gases by increasing RCS pressure through the use of the pressurizer backup heaters and increased charging flow. If this step is successful in condensing the gas volume in the vessel head (as indicated by a return to normal readings in those parameters used to determine the presence of the gases) then return to the appropriate operating instruction.

CAUTION: Increased charging flow with condensible gases in the RCS may result in a decreasing pressurizer level. If pressurizer level decreases to less than 20% of span, then attempt to restore level by continuing the charging flow or manually starting safety injection pumps. If level cannot be restored, then manually initiate safety injection and proceed to EOI-0, Immediate Actions and Diagnostics.

3. In preparation for venting, isolate the containment purge and exhaust system and the pressure vacuum relief line and start all available containment air circulation equipment.

4. Increase the RCS sub-cooling to (insert plant specific value which is 50°F above the value which is the sum of the errors for the temperature measurement system used, and for the pressure measurement system translated into temperature using the saturation tables) by either initiating an RCS pressurization or by dumping steam from the non-faulted steam generators.

5. If required, perform the actions of Appendix B to determine the maximum allowable time period for venting (only for plants which vent directly to containment).

*6. Isolate letdown and initiate an RCS makeup by the chemical volume and control system to increase pressurizer level to greater than 50% of span.

*7. If not already performed, manually block the low pressure SI initiation if the permissive is energized.

CAUTION: The venting operation may result in pressure decreasing below the SI setpoint. Action should be taken to manually block the automatic SI signal when the permissive is energized.

*8. Increase charging flow to maximum to limit the pressurizer pressure and level decrease during the venting period.

NOTE: Observe the pressurizer level trend during the venting and, from the following conditions, determine the probable status of the reactor coolant system.

a) Increasing pressurizer level - Gaseous voids exist in the RCS other than the reactor vessel head or pressurizer.

b) Constant pressurizer level - No significant gaseous voids exist in the reactor coolant system.

c) Decreasing pressurizer level - Gaseous void exists in the reactor vessel head.

9. Open the vent isolation valves in one head vent flow path.

NOTE: If one or both valves fail to open, close both valves and open the isolation valve in the parallel flow path.

10. Close both vent isolation valves when:

a) Reactor vessel level indication stabilizes,
OR
b) The time period determined in Step 5 is met,
OR
c) Pressurizer pressure decreases by 200 psi,
OR
d) Pressurizer level decreases below 20 percent of span
OR
e) Reactor coolant sub-cooling decreases below (insert plant specific value which is the sum of the errors for the temperature measurement system used, and for the pressure measurement system translated into temperature using the saturation tables).
OR
f) The reactor vessel head is refilled as indicated by a decrease in the rate of a depressurization or a change in the rate of the pressurizer level trend.

CAUTION: If during the venting period, a loss of reactor coolant pump operation occurs, continue the venting and allow natural circulation to establish itself.

*11. Re-establish normal charging and letdown to maintain the pressurizer water level in the operating range.

*12. Evaluate the response of the pressurizer level trend to determine if a gas bubble existed in the vessel head. If a gas bubble existed and the venting was terminated prior to the vessel head being completely refilled, then return to Step 4.

NOTE: If multiple venting operations are required and the containment hydrogen concentration is equal to or greater than 3 volume percent, the provisions must be made to remove or reduce the volume of hydrogen from the containment prior to re-opening the reactor vessel head vent.

13. Return to the appropriate operating instruction following the successful completion of the venting of the reactor vessel head.

APPENDIX "A"

RV HEAD VENT GUIDELINE

RCS GASEOUS VOID

DETECTION AND SIZING

1. Achieve a constant pressurizer level and pressure condition.

2. Place the RCS wide range or pressurizer pressure and the pressurizer level on trend recorders. The scale should be 150 psig pressure and 10% of span for level.

3. Record the following parameters.

RCS Pressure	=	_____PSI
PZR Level	=	_____%
Charging Rate	=	_____GPM
Seal Injection Flow	=	_____GPM
Seal Leakoff Low	=	_____GPM
Time	=	_____

4. Isolate the RCS letdown flow, turn off all pressurizer heaters, and terminate the pressurizer spray by placing the spray control in manual and zeroing the demand signal.

5. Allow the RCS charging flow to either increase RCS pressure 100 psi or increase pressurizer level 5% of span.

6. Record the RCS pressure, pressurizer level and time.

RCS Pressure	=	_____PSI
PZR Level	=	_____%
Time	=	_____

7. Reinitiate RCS letdown flow and restore normal pressurizer pressure and level control.

8. Calculate the initial and final pressurizer vapor space volumes.

Initial Vapor Volume = (1-PZR Level % X Total Cylindrical PZR Volume FT^3) + (Upper Spherical Volume FT^3)

= _____ FT^3

Final Vapor Volume = (Initial Volume) - (Δ PZR Level X Total Cylindrical Volume)

= _____ FT^3

9. Determine the total charged volume into the RCS.

Charged Volume = (Charging + Seal Injection - Seal Leakoff GPM) X

$$(\text{Time}) \times \left(\frac{1}{7.45 \frac{GPM}{FT^3}}\right)$$

= _____ FT^3

10. Determine the expected pressurizer level change.

$$\text{Expected } \Delta \text{ level} = (\text{Charging Volume } FT^3) \times (\text{Time}) \times \left(\frac{100\%}{\text{Total PZR Volume } FT^3}\right)$$

= _____ %

11. If the actual pressurizer level change is less than the expected level change then a gaseous void exists in the reactor coolant system. Perform the following step to determine the volume of the RCS void.

12. The initial and final RCS gaseous void volumes can be calculated from the following equations.

$$\text{Initial RCS Void} = \frac{(\text{Initial Vapor Volume}) - (\text{Final Vapor Volume}) - (\text{Charged Volu}}{\left(1 - \frac{\text{Initial Pressure}}{\text{Final Pressure}}\right)}$$

= _____ FT^3

$$\text{Final RCS Void} = \frac{(\text{Initial RCS Void}) \times (\text{Initial Pressure})}{(\text{Final Pressure})}$$

= _____ FT^3

APPENDIX "B"

RV HEAD VENT GUIDELINE

VENTING TIME PERIOD

1. Convert the containment free-volume to containment volume at standard temperature and pressure conditions.

$$\text{Cont. Volume (STP)} = (\text{Cont. Volume } FT^3) \times \left(\frac{\text{Cont. Pressure}^{**}}{14.7 \text{ PSIA}}\right) \times \left(\frac{492°R}{\text{Cont. Temp.}^{*}}\right)$$

$$= \underline{\qquad\qquad} FT^3$$

* Temperature in degrees Rankine (°F + 460)

**If containment pressure has increased above 14.7 psia then use 14.7 psig as pressure for conservatism.

2. Determine the containment hydrogen concentration in volume percent units.

<u>NOTE</u>: The containment hydrogen concentration will be insigificant if there has been no leakage from the RCS to the containment.

3. Calculate the maximum hydrogen volume that can be vented to the containment which will result in a containment hydrogen concentration of less than or equal to 3 volume percent.

$$\text{Maximum } H_2 \text{ Volume to be Vented} = \frac{(3.0\% - \text{Cont. } H_2 \text{ Concentration } \%) \times (\text{Cont. Volume [STP]})}{100\%}$$

$$= \underline{\qquad\qquad}\%$$

4. From Curve #1 (RCS Pressure vs. H_2 Flow Rate) determine the allowable venting period which will limit the containment hydrogen concentration to 3 volume percent.

$$\text{Venting Period} = \frac{\text{Max. } H_2 \text{ Vented}}{H_2 \text{ Flow Rate}} \text{ (From Step 3)}$$

$$= \underline{\qquad\qquad} \text{ Mins.}$$

13.5.1.10 Control of Special Processes During Operations Procedures

These procedures assure that special processes are accomplished under controlled conditions in accordance with applicable codes, standards, specifications, criteria and other special requirements using qualified personnel and procedures. 3

13.5.1.11 Nonconformance Control/Deficiency Reporting Procedures

These procedures provide for control of items, services or activities which do not conform to requirements. These procedures include instructions for identification, documentation, segregation, notification of affected organizations and method of disposition of such items, services or activities. 3

13.5.1.12 Test Control Procedures

These procedures assure that testing required to demonstrate that an item will perform satisfactorily in service is accomplished properly. Test procedures incorporate or reference the requirements and acceptance limits contained in applicable design documents. These test procedures may include preoperational tests, initial operational phase tests, surveillance tests and tests during design, fabrication and construction activities associated with plant maintenance and modification. 3

13.5.1.13 Feedback of Operating Experience

In accordance with NUREG 0737, item I.C.5, a program will be established for evaluating operating plant experience and providing the results of the evaluations, as necessary, to pertinent plant personnel. This program will primarily be performed as a function of the Shift Technical Advisor Group. The services of "Industry Groups" such as INPO will be utilized to the extent possible in the performance of this function. 23 24

Insert on Next Page → 13,5,1,14

13.5.2 CONTROL ROOM OPERATING PROCEDURES

Control room operating procedures are those procedures that are performed by the licensed Control Room Operator or under his direction

13.5.1.14 ECCS Outages

In accordance with NUREG 0737, item II k 3.17, a program has been established using existing plant procedures for data collection for determining ECCS outage times. A plant procedure for removal and restoration of station equipment provides measures for data collection. The ECCS data taken by this procedure will be reviewed by appropriate plant personnel to determine if improvements to availability of ECCS is needed.

1. Dilution During Refueling

An uncontrolled boron dilution accident based on a failure in the primary water makeup system cannot occur during refueling. This accident is prevented by administrative controls which isolate the RCS from the potential source of unborated water. 6

Valves ~~PCV-113B, PCV-160A,~~ 8430, 8454, 8441, and 8439 will be locked closed during refueling operations. These valves will block the flow paths which could allow unborated makeup water to reach the RCS. Any makeup which is required during refueling will be ~~borated water supplied from the refueling water storage tank by the charging pumps.~~ ADDED TO THE REACTOR COOLANT SYSTEM BY UNLOCKING THESE VALVES AND INITIATING THE REQUIRED BLENDED MAKEUP WATER FLOW. AFTER THE REQUIRED VOLUME OF BLENDED MAKEUP FLOW HAS BEEN ADDED, THESE VALVES WILL AGAIN BE LOCKED CLOSED. AN ALTERNATE SOURCE OF BORATED WATER THAT COULD BE USED IS FROM THE REFUELING WATER STORAGE TANK TO THE CHARGING PUMP SUCTION. 5 6

The most limiting alternate source of uncontrolled boron dilution would be the inadvertent opening of a valve in the boron thermal regeneration system (BTRS). For this case, highly borated RCS water is depleted of boron as it passes through the BTRS and is returned via the volume control tank. The following conditions are assumed for an uncontrolled boron dilution during refueling. 6

Technical Specifications require the reactor to be borated to at least 2,000 ppm and shutdown by at least 5.0 percent $\Delta k/k$ at refueling. The maximum boron concentration to lose all shutdown margin is very conservatively estimated to be 1,500 ppm. 23

Dilution flow is assumed to be the maximum capacity of the BTRS (120 gpm) with 0 ppm water returning to the RCS. This is assumed although normally this system is not operated at refueling conditions. 6

Mixing of the reactor coolant is accomplished by the operating of one residual heat removal pump.

A minimum water volume (3000 ft^3) in the RCS is used. This is a conservative estimate of the minimum volume of the RCS for residual heat removal system operation. 23

2. Dilution During Cold Shutdown

Technical Specifications require the reactor to be shutdown by at least 1.0 percent Δk/k during cold shutdown. The minimum boron concentration required to meet this shutdown margin is conservatively estimated to be 1672 ppm. If the reactor is in cold shutdown and on the residual heat removal system with RCS piping filled and vented, the following conditions are assumed for an uncontrolled boron dilution. Dilution flow is assumed to be a maximum of 150 gpm, which is the capability of one primary water makeup pump to deliver unborated water to the RCS. Mixing of the reactor coolant is accomplished by the operation of one residual heat removal pump.

A minimum volume of 4705 ft^3 in the reactor coolant system is used. This corresponds to the active volume of the reactor coolant system minus the pressurizer volume, while on the residual heat removal system.

23 If the reactor is in cold shutdown and the RCS water level is drained down to reactor vessel mid-nozzle while on RHR, an inadvertent dilution is prevented by administrative controls which isolate the RCS from the potential source of unborated water. Valves ~~MOV-1115~~, ~~MOV-1051~~, 8430, 8454, 8441, and 8439 will be locked closed during operations in these conditions. These valves block all flow paths that could allow unborated makeup water to reach the RCS. Any makeup which is required will be ADDED TO THE REACTOR COOLANT SYSTEM BY ~~borated water supplied from the RWST by the low head safety injection pumps~~. UNLOCKING THESE VALVES AND INITIATING THE REQUIRED BLENDED MAKEUP WATER FLOW. AFTER THE REQUIRED VOLUME OF BLENDED MAKEUP WATER FLOW HAS BEEN ADDED, THESE VALVES WILL AGAIN BE LOCKED CLOSED. AN ALTERNATE SOURCE OF BORATED WATER THAT MAY BE USED IS FROM THE REFUELING WATER STORAGE TANK TO THE CHARGING PUMP SUCTION.

3. Dilution During Hot Standby

Technical Specifications require the reactor to be shutdown by at least 1.77% Δk/k during hot standby. The minimum boron concentration required to meet this shutdown margin is very conservatively estimated to be 1556 gpm. The following conditions are assumed for a continuous boron dilution during hot standby: