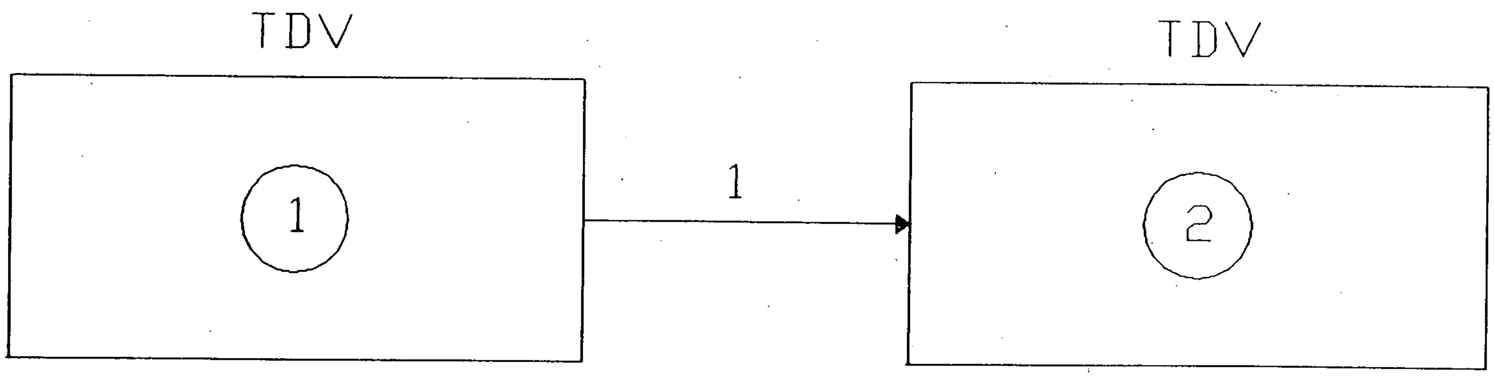
Fig. 1 RETRAN Nodal Scheme For Calculating Critical Flow Rate



(1) Time-dependent volume (TDV) is a boundary volume in which the fluid conditions are specified as a function of time.

(2) Critical flow model, flow area and discharge coefficient are specified at Junction 1.

Fig. 2 SONGS 1 RETRAN OMS Analysis Model

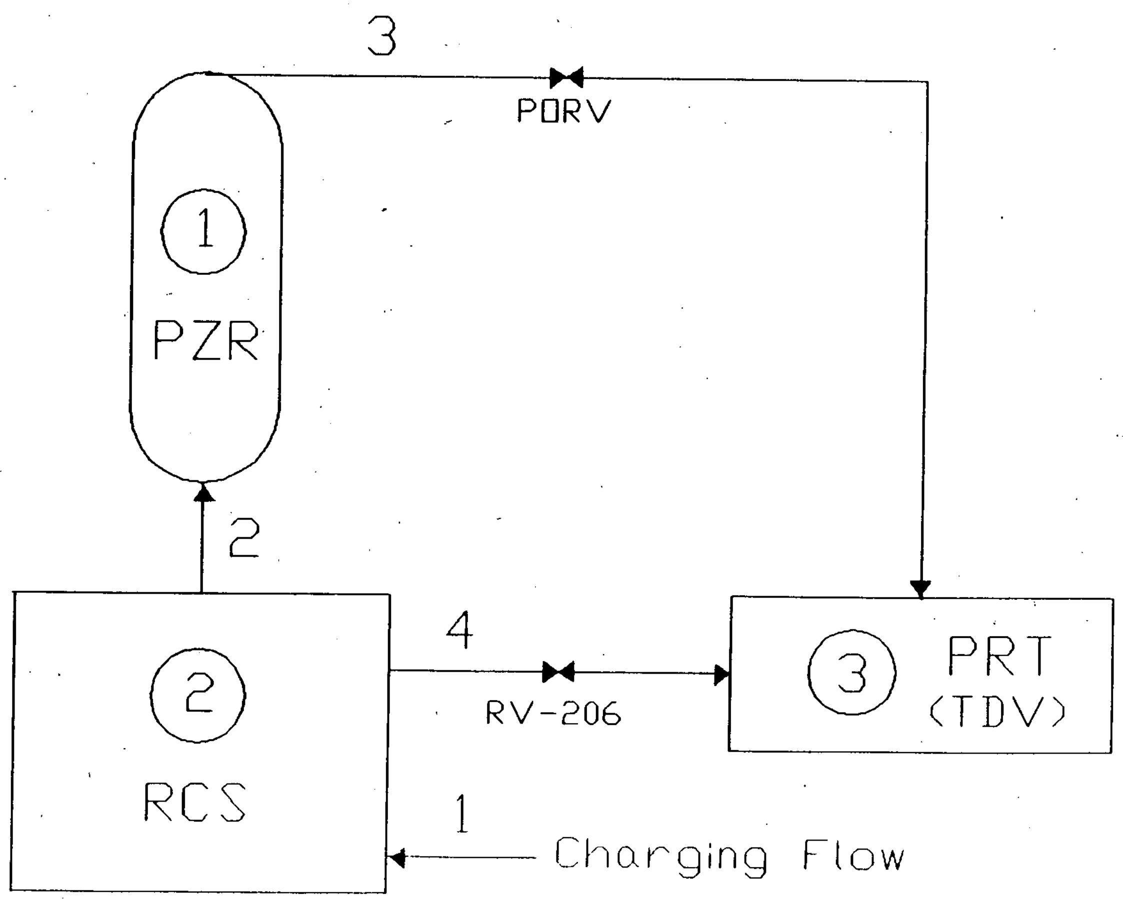
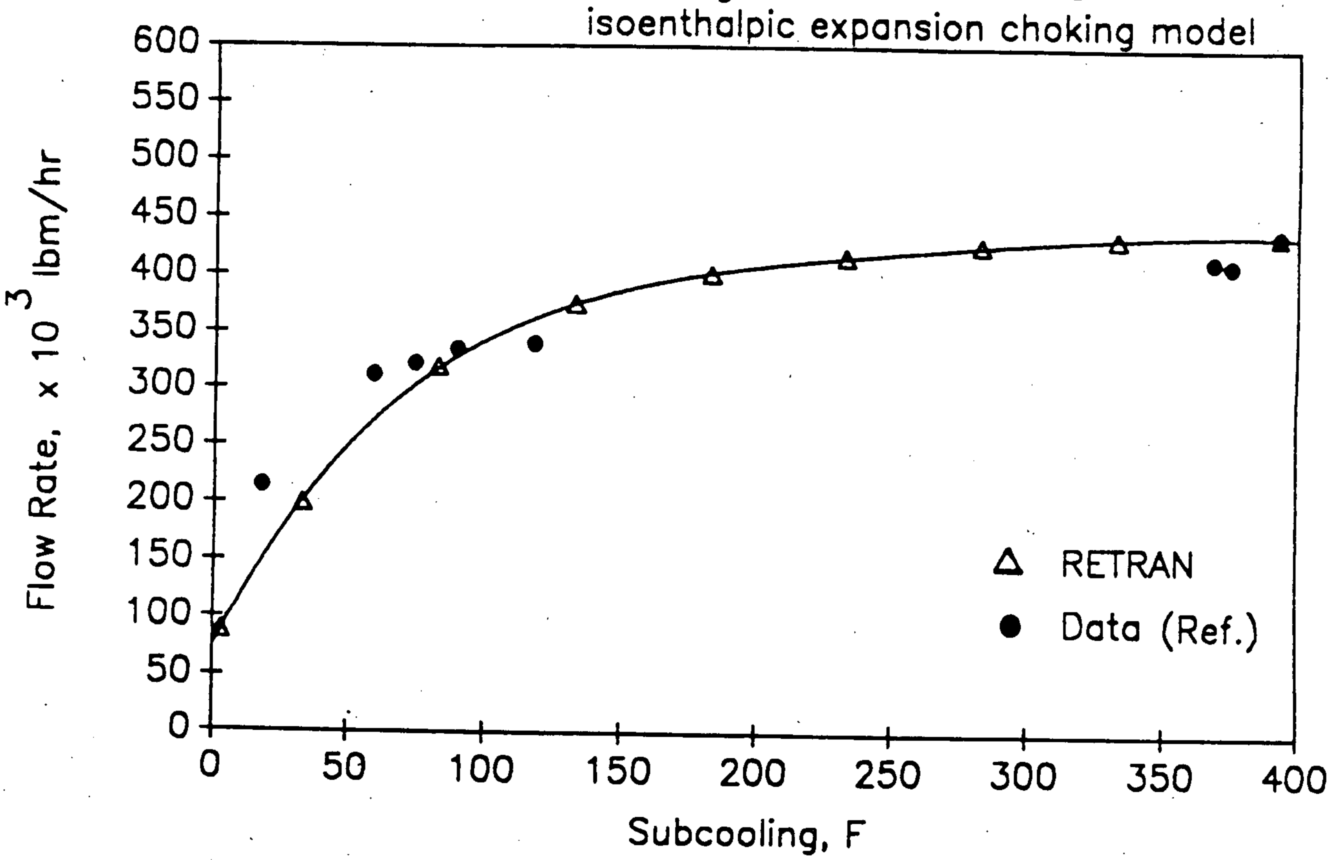


Fig. 3 SONGS 1 RV-206 Water Flow Discharge Versus Temperature

RETRAN Input : upstream pressure = 580 psia
valve flow area = 1.28 sq. inch
discharge coefficient = 0.75
isoenthalpic expansion choking model



Ref.: Table 1, Flow Testing of Crosby 2 1/2 J4 JB-35-TD, Type E Relief Valve, Test Report Number 4637, July 10, 1990

VIII. ATTACHMENT

Attachment 1 - Listing of RETRAN Input Data for Calculating Critical Flow Rate

Attachment 2 - Listing of RETRAN Input Data for SONGS 1 OMS Analysis Model

STING OFINPUT DATA FOR CASE 1

```
= RETRAN CALCULATION OF CHOKED FLOW
*
*
*        _______________                 ____________
*        1               1               1            1
*        1    TDV #1     1     JUN 1     1  TDV #2    1
*        1  P = 580 PSIA 1-------------->1            1
*        1  T =  90 F    1   C  = 0.75   1            1
*        1_______________1    D          1____________1
*        ( T   = 482.6 F)    A  = 1.28 SQ.IN
*           SAT               C
*
*
*          ***************************************
*          *                                     *
*          *     CODE VERSION: RETRAN02-MOD4     *
*          *                                     *
*          ***************************************
*
*
*
************* PROBLEM CONTROL AND DESCRIPTION DATA - 01000Y ************
*
*
*----Y------1-----2-----3-----4-----5-----6-----7-----8-----9----10----
*
010001     0     6     2     1     2     0     2     1     0     0
*
*
*----------11----12----13----14----15----16----17----18----19----20----
*
010002     0     0     0     0     0     0     0     0     0     0
*
*
*----------21----22----23----24----25----26----27----28----29----30----
*
010003     0     0     0     0     1     0     0     0     0     0
*
*
*----------31----32----33----34----35----36----37----38----39----40----
*
010004     0     0     1     0     0     0     0     0     0     0
*
*
*
******************* MINOR EDIT VARIABLES - 02000Y *******************
*
*    Y
*
020001  NP**,  1   ICHK,  1   PRES,  1   TEMP,  1   PRES,  2   TEMP,  2
*
*
********************** TIME STEPS DATA - 03XXX0 *********************
*
*
*
*        NMIN    NMAJ    NDMP   NCHK    DELTM     DTMIN     TLAST
*
```

ATTACHMENT 1

```
030010         20     100      9999     0      0.01     0.0       10.0
030020         50     500      9999     0      0.01     0.0     1000.0
*
*
*
********************** TRIP CONTROLS - 04XXX0 ************************
*
*
*
*        TRIP  SIG
*4XXX0    ID    ID   IX1   IX2    SETPT   DELAY
*
040010    1     1     0     0      1.0     0.0  * PROBLEM RUNNING TIME
*
*
******************** VOLUME DATA - 05XXXY (Y=1) **********************
*
*
*
*          1  2      3     4      5      6       7       8       9       10
*      Y  IB IR      P     T      X     VOL     ZVOL     ZM    FLOWL    FLOWA
*                         (H)
050011    0  1     0.0   0.0    0.0  1000.0    25.0    25.0    50.0    10.0
050021    0  2     0.0   0.0    0.0  1000.0    25.0    25.0    50.0    10.0
*                                   <---------  ASSUMPTION -------->
*
*-----  VOLUME DATA - 05XXXY (Y=2)
*
*               11          12         13        14       15         16
*     Y       DIAMV        ELEV       INEQ     VRAIN    VLHTC       MESH
*
050012        10.0         0.0
050022        10.0         0.0
             (ASSUMPTION)
*
*
*
********************** TIME-DEPENDENT VOLUMES (07XXYY) *****************
*
*
*           N         TIME          PRES        TEMP      AVG.      MIXL
*                    (SEC)         (PSIA)       (F)        X        (FT)
*
070101      2          0.0         580.0        90.0      0.0       25.0
070102              1.0E+6         580.0        90.0      0.0       25.0
*
070201      2          0.0          25.0        70.0      0.0       25.0
070202              1.0E+6          25.0        70.0      0.0       25.0
*
*
*
********************** JUNCTION DATA - 08XXXY (Y=1) ********************
*
*
*
*          1    2   3   4      5          6          7         8         9         10
* XXXY    VI   VO  IP  IV     WP        AJUN       ZJUN     INERTA     FJUNF     FJUNR
*
080011     1    2   0   0    0.0   8.8889E-3    25.0       1.0       0.0       0.0
*                                 (1.28 IN*IN) <------ ASSUMPTION ------->
```

```
*
****** JUNCTION DATA - 08XXXY (Y=2)
*
*
*         11    12     13    14    15       16     17     18     19      20   21
*     Y JVERT CHOK   JCA   MIX  DIAMJ    CNTR   REGM   2PHS   ANGL   INDEX  ISP
*
080012    0     1      0     0    0.0      0.75    0     -1     0.0      0    0
*
*       AJUN * CNTR = 8.8889E-3 * 0.75
*                   = 0.96 SQ. INCH
*
*************************** END OF DATA *********************************
.
```

```
= RETRAN ANALYSIS OF SONGS 1 OMS TRANSIENTS - HEAT ADDITION
*
*     * * * * * * * * * * * * * * * * * * * * * * * * * * *
*     *                                                   *
*     *      BASIC MODEL - NEW RV206 FLOW (469 GPM)       *
*     *                    CHARGING FLOW = 0.0 GPM        *
*     *                                   (420 GPM)       *
*     *                    DELTA T =   5 F  OR  30 MW     *
*     * * * * * * * * * * * * * * * * * * * * * * * * * * *
*
* PZR (NODE 1):                                  RETRAN NODING DIAGRAM
*     VOLUME,  FT**3 = 1300                                    ________
*     PRESSURE, PSIA =  415                                   1        1
*     TEMPERATURE, F =  SAT                   ________ JUN 3  1 NODE 3 1
*                                            1        1--->&lt;--->1 (PRT)  1
* RCS (NODE 2):                              1 NODE 1 1 (PORV)  1________1
*     VOLUME,  FT**3 = 5450                  1 (PZR)  1                1
*     PRESSURE, PSIA = 415                   1________1                1
*     TEMPERATURE, F = 140                       1                     1 JUN 4
*                                                1 JUN 2               1
* PRT (NODE 3) - TDV:                      ______1_______   (RV-206)   1
*     PRESSURE, PSIA = 14.7               1             1---->&lt;-----1
*     TEMPERATURE, F =   70               1   NODE 2    1
*                                         1   (RCS)     1        (CHARGING)
* PORV CHARACTERISTICS (JUN 3):           1_____________1&lt;------ JUN 1
*     LIFT SETPOINT = 525 PSIA
*     DELAY         = 0.6 SECOND
*     STROKE TIME   = 1.9 SECONDS
*     CD * AC       = 0.583
*     FLOW (LIQUID) = ISOENTHALPIC MODEL
*
* RV 206 CHARACTERISTICS (JUN 4):
*     LIFT SETPOINT    = 515 PSIG/530 PSIA (AT THE VALVE)
*     VALVE FLOW AREA = 1.28 INCH**2
*     DISCHARGE COEF. = 0.42
*     FULL CAPACITY    = 469 GPM
*     ISOENTHALPIC EXPANSION CRITICAL MODEL
*
*        RETRAN          RV-206
*        (PSIA)         POSITION
*   --------------------------------
*          0.0             0.0
*        502.0             0.0       * 28 PSI HEAD BETWEEN PZR AND RV-206
*        503.0             0.9
*        552.0             1.0
*       1.0E+6          1.0E+6
*
* CHARGING FLOW (JUN 1):
*     FLOWRATE      = 420 GPM
*     TEMPERATURE   = 145 F (MAXIMUM PRESSURIAZATION)
*
*
*
*          **************************************
*          *                                    *
*          *    CODE VERSION: RETRAN02-MOD4     *
*          *                                    *
*          **************************************
```

ATTACHMENT 2

```
*
*
*
************* PROBLEM CONTROL AND DESCRIPTION DATA - 01000Y ************
*
*
*----Y------1-----2-----3-----4-----5-----6-----7-----8-----9----10----
*
010001      0    -11    2     5     3     0     1     4     0     2
*
*
*----------11----12----13----14----15----16----17----18----19----20----
*
010002      3     1     0     0     0     0     1     0     0     0
*
*
*----------21----22----23----24----25----26----27----28----29----30----
*
010003      0     0     0     0     1     0     0     0     1     0
*
*
*----------31----32----33----34----35----36----37----38----39----40----
*
010004      0     0     1     0     0     0     0     0     0     0
*
*
*
******************** MINOR EDIT VARIABLES - 02000Y ********************
*
*    Y
*
020001  PRES,  1   TEMP,  1   PRES,  2   TEMP,  2   PRES,  3   TEMP,  3
020002  MP**,  1   MP**,  3   MP**,  4   COUT, -1   COUT, -2
*
*       COUT, -2 = HEAT ADDITION, MEGAWATTS
*
*********************** TIME STEPS DATA - 03XXX0 *********************
*
*
*
*          NMIN    NMAJ    NDMP   NCHK    DELTM    DTMIN    TLAST
*
030010      10     100     9000    0      0.01      0.0      30.0
030020      50     100     9000    0      0.01      0.0    1000.0
*
*
*
*********************** TRIP CONTROLS - 04XXX0 ***********************
*
*
*
*       TRIP   SIG
*4XXX0   ID     ID   IX1  IX2   SETPT   DELAY
*
040010   1      1     0    0     30.0     0.0  * PROBLEM RUNNING TIME
*
040020   2      1     0    0      0.0     0.0  * FOR CHARGING FLOW
040020   2      1     0    0      0.0  1.0E+6  * FOR CHARGING FLOW
*
040030   3      4     1    0    525.0     0.6  * PORV LIFT SETPOINT
```

*** THIS CARD IS A REPLACEMENT CARD. ***

```
*
040040    4    4    1    0    502.0    0.1  * RV-206 LIFT SETPOINT
*
040050    5    1    0    0      0.0    0.0  * FOR NON-CONDUCTING HX
*
*
*
******************* VOLUME DATA - 05XXXY (Y=1) ***********************
*
*
*
*          1  2     3      4       5       6       7       8       9       10
*     Y  IB IR      P      T       X      VOL    ZVOL     ZM    FLOWL    FLOWA
*                         (H)
050011   0  0   415.0    0.0    -1.0   1300.0    50.0    50.0    50.0    50.0
050021   0  0   415.0  108.98    0.0   5450,0    50.0    50.0    50.0    50.0
*                      (140 F)                  <----- ASSUMPTION ---->
050031   0  1    14.7   70.0     0.5   2.0E+6   100.0     0.0   100.0  1.0E+6
*
*-----  VOLUME DATA - 05XXXY (Y=2)
*
*              11          12        13       14       15       16
*     Y       DIAMV       ELEV      INEQ     VRAIN    VLHTC     MESH
*
050012        20.0        0.0
050022        20.0        0.0
*            (ASSUMPTION)
050032        50.0        0.0
*
*
*
**************** TIME-DEPENDENT VOLUME 07XXYY *********************
*
*
*          IRIN      TIME       PRES       TEMP      AVG.      MIXL
* XXYY               (SEC)     (PSIA)      (F)        X        (FT)
*
070101      2         0.0       25.0       70.0      0.5       0.0
070102               1.0E+6     25.0       70.0      0.5       0.0
*
*
*********************** JUNCTION DATA - 08XXXY (Y=1) *****************
*
*
*
*        1   2   3  4      5        6          7         8         9         10
* XXXY  VI  VO  IP IV     WP       AJUN       ZJUN     INERTA    FJUNF     FJUNR
*
080011   0   2   1  0     0.0      1.0        1.0       0.0       0.0       0.0
080021   2   1   0  0     0.0     1.0E+6      0.0     1.0E-6      0.0       0.0
080031   1   3   0  1     0.0      1.0        0.0     1.0E-6      0.0       0.0
080041   2   3   0  2     0.0    8.8889E-3    1.0     1.0E-6      0.0       0.0
*                          (1.28 SQ. INCH)
*
****** JUNCTION DATA - 08XXXY (Y=2)
*
*
*        11    12    13    14    15     16     17     18     19     20   21
*     Y JVERT CHOK  JCA   MIX   DIAMJ  CNTR   REGM   2PHS   ANGL  INDEX ISP
*
```

```
 080012   0   -1    0    0    0.0     0.0      0    -1   0.0    0   0
 080022   0   -1    0    0    0.0     0.0      0    -1   0.0    0   0
 080032   0    1    0    0    0.0  4.0486E-3   0    -1   0.0    0   0
 *        NOTE: CNTR * AJUN = 4.0486E-3 FT**2 ( = 0.583 INCH**2)
 *
 080042   0    1    0    0    0.0     0.42     0    -1   0.0    0   0
 *        NOTE: CNTR * AJUN = 3.7333E-3 FT**2 ( = 0.538 INCH**2)
 *
 *
 **********************  VALVE DATA - 11XXX0  ***********************
 *
 *
 *       TRIP
 * XXX0   ID     IACV     IAC2     PCV     CV1     CV2     CV3
 *
 110010  -3       1        0        0       0       0       0
 *
 110020  1000    -3        0        0       0       0       0
 *
 *
 **********************  GENERAL DATA - 12XXYY  *********************
 *
 *
 *                    TIME        NORM.
 * XXYY       N       (SEC)       CURVE
 *
 120101      -3        0.0         0.0        * PORV
 120102                1.9         1.0
 120103              1.0E+6        1.0
 *
 *            N       PRES.       NORM.
 *                    (PSIA)      CURVE
 *
 120201      -5        0.0         0.0        * RV-206
 120202              502.0         0.0
 120203              503.0         0.9
 120204              552.0         1.0
 120205              1.0E+6        1.0
 *
 *
 *                    TIME       NORM. HEAT
 *            N       (SEC)      ADDITION
 *
 120301      -6     -1.0E+6       0.0
 120302                0.0        0.0
 120303                0.1        1.0
 120304               20.0        1.0
 120305               20.1        0.0
 120306              1.0E+6       0.0
 *
 *
 ************************* FILL TABLE - 13XXYY **********************
 *
 *
 *
 *        ---------- TABLE 1 (JUN 1 - CHARGING FLOW) -----------
 *
 *
 *                TRIP    JX          TIME       FLUX        H       PSIA
 * XXYY     N      ID   (TIME)  JY    (SEC)   (GPM/FT**2)
```

```
*
130101   -2     2      0     1    0.0         420.0      115.0   1000.0
130102                               1.0E+6        420.0      115.0   1000.0
*                                                         (145 F)
*
*
*
********************  HEAT EXCHANGER - 21XXYY ***    ****************
*
*
*
*          IHTX      ITHTXQ     JVOL     IHTYPE      W5       W6
*       (CNTL BLK)
210101      -2         1000       2         7        0.0      0.0
*
*
*
****************  CONTROL SYSTEM MODELING - 70YXXX  *****************
*
*
*           NO. OF CNTL          NO. OF             MAX TIME-
*           INPUT BLOCKS      CONTROL BLOCKS        STEP SIZE
*
701000           3                  3                  0.01
*
*
*           -----  CONTROL INPUT BLOCK DEFINITION  -----
*
*
*  XXX     IDC     SYMBOL      IREG      CGAIN       CIC
*
702001      1       TIMX,        0        1.0        0.0
702002      2       TRPT,        5        1.0      1.0E+6
702003      3       PRES,        1        1.0       415.0
*
*
*              -----  CONTROL BLOCK DEFINITION  -----
*
*
*         IDC  ITYPE   INC1   INC2    GAIN     CP1    CP2    CIC     CMIN    CMAC
*
703001  -1     SUM,     1      2      1.0      1.0   -1.0  -1.0E+6
703002  -2     FNG,    -1      3    -30.0      0.0    0.0     0.0   * HEAT IN
*                                                                      (MW)
703003  -3     FNG,     3      2      1.0      0.0    0.0     0.0   * RV-206
*                                                                     POSITION
*
**************************** END OF DATA ********************************
.
```

ATTACHMENT 1

EXISTING TECHNICAL SPECIFICATIONS

3.1 REACTOR COOLANT SYSTEM

3.1.1 MAXIMUM REACTOR COOLANT ACTIVITY

APPLICABILITY: Applies to measured maximum activity in the reactor coolant system at any time.

OBJECTIVE: To limit the consequences of an accidental release of reactor coolant to the environment.

SPECIFICATION: The specific activity of the reactor coolant shall be limited to:

1. $\leq$ 1.0 µCi/gm DOSE EQUIVALENT I-131.

2. $\leq 100/\bar{E}$ µCi/gm.

ACTION:

A. With the specific activity of the reactor coolant determined to be >1 µCi/gm DOSE EQUIVALENT I-131 for more than 48 hours during one continuous time interval or >60 µCi/gm DOSE EQUIVALENT I-131 or $>100/\bar{E}$ µCi/gm, be in at least HOT STANDBY with the average temperature of the reactor coolant (T_{avg}) less than 535°F within 6 hours.

B. With the specific activity of the reactor coolant > 1.0 µCi/gm DOSE EQUIVALENT I-131 or $>100/\bar{E}$ µCi/gm, perform the sampling and analysis requirements of item 1a.4.a of Table 4.1.2 until the specific activity of the reactor coolant is restored to within its limits.

C. The provisions of Specification 3.0.4 are not applicable.

Specific Activity

BASIS:

The limitations on the specific activity of the reactor Coolant ensure that the resulting 2 hour doses at the site boundary will not exceed the guidelines of 10 CFR Part 100 following a steam generator tube rupture accident in conjunction with an assumed steady state primary-to-secondary steam generator leakage rate of 1.0 GPM.

The ACTION statement permitting POWER OPERATION to continue for limited time periods with the reactor coolant's specific activity > 1.0 μCi/gm DOSE EQUIVALENT I-131, accommodates possible iodine spiking phenomena which may occur following changes in THERMAL POWER.

Reducing T_{avg} to < 535°F prevents the release of activity should a steam generator tube rupture since the saturation pressure of the reactor coolant is below the lift pressure of the atmospheric steam relief valves. The surveillance requirements provide adequate assurance that excessive specific activity levels in the reactor coolant will be detected in sufficient time to take corrective action. Increased surveillance for performing isotopic analyses for iodine is required whenever the DOSE EQUIVALENT I-131 exceeds 1.0 μCi/gram and following a significant change in power level to monitor possible iodine spiking phenomena to assure the activity remains < 60 μCi/gm DOSE EQUIVALENT I-131.

The assumptions and results of these calculations are documented in "Safety Evaluation by the Office of Nuclear Reactor Regulation," Docket No. 50-206, dated April 1, 1977.

3.1.2 <u>OPERATIONAL COMPONENTS</u>

<u>APPLICABILITY</u>: Applies to the operating status of the reactor coolant system equipment and related equipment. For the applicable surveillance requirements, see Table 4.1.2.

<u>OBJECTIVE</u>: To identify those conditions of the reactor coolant system necessary to ensure safe reactor operation.

<u>SPECIFICATIONS</u>:

A. At least one pressurizer safety valve shall be OPERABLE or open when the reactor head is on the vessel, except for hydrostatic tests.

B. The reactor shall not be made critical or maintained critical unless both pressurizer safety valves are OPERABLE.

C. During MODES 1 and 2 and in MODE 3 with reactor trip breakers closed, all three reactor coolant loops and their associated steam generators and reactor coolant pumps shall be in operation. With less than the above required coolant loops in operation, be in at least HOT STANDBY with reactor trip breakers open within 1 hour, except as modified by Specification D below.

D. The limitations of Specification C may be suspended as follows:

1. During MODES 1 and 2, operation may be conducted with 0, 1, 2 or 3 reactor coolant pumps operating at less than 5% of RATED THERMAL POWER for purposes of conducting low power physics testing.

2. During MODES 1 and 2 and in Mode 3 with reactor trip breakers closed, operation may be conducted for less than 24 consecutive hours with one or two reactor coolant pumps operating if THERMAL POWER is less than 10% of RATED THERMAL POWER.

E. During MODE 3 with the reactor trip breakers open, the following specifications shall apply:

1. At least two of the reactor coolant loops listed below shall be OPERABLE:

a. Reactor coolant loop A and its associated steam generator and reactor coolant pump.

b. Reactor coolant loop B and its associated steam generator and reactor coolant pump.

c. Reactor coolant loop C and its associated steam generator and reactor coolant pump.

2. At least one of the above reactor coolant loops shall be in operation.*

3. With less than the above required reactor coolant loops OPERABLE, restore the required loops to OPERABLE status within 72 hours or be in HOT SHUTDOWN within the next 12 hours.

4. With no reactor coolant loop in operation, suspend all operations involving a reduction in boron concentration of the reactor coolant system and immediately initiate corrective action to return the required reactor coolant loop to operation.

F. During MODE 4, the following specifications shall apply:

1. At least two of the reactor coolant loops/RESIDUAL HEAT REMOVAL (RHR) TRAINS listed below shall be OPERABLE:

   a. Reactor coolant loop A and its associated steam generator and reactor coolant pump.

   b. Reactor coolant loop B and its associated steam generator and reactor coolant pump.

   c. Reactor coolant loop C and its associated steam generator and reactor coolant pump.

   d. Residual heat removal (RHR) pump G-14A and one associated RHR TRAIN.

   e. Residual heat removal (RHR) pump G-14B and one associated RHR TRAIN.

2. At least one of the above loops/trains shall be in operation.**

* All reactor coolant pumps may be de-energized for up to one hour provided (a) no operations are permitted that would cause dilution of the reactor coolant system boron concentration, and (b) core outlet temperature is maintained at least 40°F below saturation temperature.

** All reactor coolant pumps and residual heat removal pumps may be deenergized for up to one hour provided (a) no operations are permitted that would cause dilution of the reactor coolant system boron concentration, and (b) core outlet temperature is maintained at least 40°F below saturation temperature.

3. With less than the above required loops/trains operable immediately initiate corrective action to return the required loops/trains to operable status as soon as possible; if the remaining operable loop/train is an RHR train, be in Cold Shutdown within 24 hours.

4. With no loop or train in operation, suspend all operations involving a reduction in boron concentration of the reactor coolant system and immediately initiate corrective action to return one required loop or train to operation.

G. During Mode 5 with reactor coolant loops filled, the following specifications shall apply:

1. At least one residual heat removal (RHR) train shall be OPERABLE and in operation*, and either

   a. One additional RHR train shall be operable,** or

   b. The secondary side water level of at least two steam generators shall be greater than or equal to 256 inches (wide range).

2. With less than the above required loops/trains operable, or with less than the required steam generator level, immediately initiate corrective action to return the required loops/trains to operable status or to restore the required level as soon as possible.

3. With no RHR train in operation, suspend all operations involving a reduction in boron concentration of the reactor coolant system and immediately initiate corrective action to return the required RHR train to operation.

* The RHR pump may be de-energized for up to one hour provided (a) no operations are permitted that would cause dilution of the reactor coolant system boron concentration, and (b) core outlet temperature is maintained at least 40•F below saturation temperature.

** One RHR train may be inoperable for up to 2 hours for surveillance testing, provided the other RHR train is operable and in operation.

H. During MODE 5 with reactor coolant loops not filled, the following specifications shall apply:

1. Two RESIDUAL HEAT REMOVAL (RHR) TRAINS shall be OPERABLE* and at least one RHR TRAIN shall be in operation**.

2. With less than the above required RHR TRAINS OPERABLE, immediately initiate corrective action to return the required RHR TRAINS to operable status as soon as possible.

3. With no RHR TRAIN in operation, suspend all operations involving a reduction in boron concentration of the reactor coolant system and immediately initiate corrective action to return the required RHR TRAIN to operation.

I. A reactor coolant pump shall not be started with the RCS pressure $\leq$ 400 psig unless:

1. the pressurizer water level is less than 80%, or

2. the potential for having developed reactor coolant system temperature gradients has been evaluated.

BASIS: One pressurizer safety valve is sufficient to prevent over-pressurizing when the reactor is subcritical, since its relieving capacity is greater than that required by the sum of the available heat sources, i.e., residual heat, pump energy and pressurizer heaters.

Prior to reducing boron concentration by dilution with make up water either a reactor coolant pump or a residual heat removal pump is specified to be in operation in order to provide effective mixing. During boron injection, the operation of a pump, although desirable, is not essential. The boron is injected into an inlet leg of the reactor coolant loop. Thermal circulation which exists whenever there is residual heat in the core and the reactor coolant system is filled and vented, will cause the boron to flow to the core.

* One RHR TRAIN may be inoperable for up to 2 hours for surveillance testing provided the other RHR TRAIN is operable and in operation.

** The RHR pump may be de-energized for up to one hour provided (a) no operations are permitted that would cause dilution of the reactor coolant system boron concentration, and (b) core outlet temperature is maintained at least 40°F below saturation temperature.

Lack of further mixing cannot result in areas of reduced boron concentration within the core. Prior to criticality the two pressurizer safety relief valves are specified in service in order to conform to the system relief capabilities.[1]

The plant is designed to have all three reactor coolant loops operational during normal power operation (MODES 1 and 2). Under these conditions, the DNB ratio will not drop below 1.30 after a loss of flow with a reactor trip.[2][3] With one reactor coolant loop not in operation, this specification requires that the plant be in at least HOT STANDBY with reactor trip breakers open within one hour (for the significance of the trip breaker position, see below). However, exception is taken whenever reactor power is less than 10% of RATED THERMAL POWER. Heat transfer analyses show that reactor heat equivalent to 8% of RATED THERMAL POWER can be removed with natural circulation only; hence, for up to 24 hours the specified upper limit of 10% of RATED THERMAL POWER with 1 or 2 reactor coolant pumps operating provides a substantial safety factor.

In MODES other than MODES 1 and 2, functional redundancy in the core heat removal methods (not necessarily system redundancy) is specified to satisfy single failure considerations. Functional redundancy, as applied to the San Onofre Unit 1 power plant, includes use of diverse heat removal methods. Furthermore, single failure considerations apply only to active components.

For operation in MODE 3 under all design basis conditions, it has been determined that one reactor coolant (RC) loop generally provides the required decay heat removal capability, the only exception to this being the control rod bank withdrawal from subcritical accident, when the DNB design basis may not be met. Since power to the gripper and lift coils of the control rod drive mechanism is carried through two reactor trip circuit breakers connected in series with the coils, both breakers must be manually closed before any control rod motion out of the core can take place. In light of this design feature, these Technical Specifications require that all three RC loops be in operation in MODE 3 if the reactor trip breakers are closed. Whenever the reactor trip breakers are open, the design feature would prevent any control rod motion, even though single failure considerations* require that at least two loops be operable. For the same reasons and subject to the same limitations that are stated in the preceding paragraph, exception is taken whenever reactor power is less than 10% of RATED THERMAL POWER.

*Single failure considerations apply to active components.

In MODES 4 and 5, the Technical Specifications permit functional redundancy in the core heat removal methods (not necessarily system redundancy) to satisfy single failure considerations. Functional redundancy, as applied to the San Onofre Unit 1 power plant, includes use of diverse heat removal methods.

In MODE 4 and MODE 5 (reactor coolant loops filled), a single reactor coolant loop or RHR TRAIN provides sufficient capability for removing decay heat; but single failure considerations* require that at least two methods (either RCS loop or RHR TRAIN) be OPERABLE.

In MODE 5 (reactor coolant loops not filled), a single RHR TRAIN provides sufficient heat removal capability for removing decay heat; but single failure considerations,* and the unavailability of any of the steam generators as a heat removing component, require that at least two RHR TRAINS be OPERABLE.

The operation of one reactor coolant pump or one residual heat removal pump provides adequate flow to ensure mixing, prevent stratification and produce gradual reactivity changes during boron concentration reductions in the Reactor Coolant System. The reactivity change rate associated with boron reduction will, therefore, be within the capability of operator recognition and control[4].

The limitation on reactor coolant pump operation with the RCS pressure $\leq$ 400 psig ensures that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50[5]. A pressurizer water level of less than 80% ensures that the start of a reactor coolant pump, with a temperature differential of 100°F will not result in 10 CFR Part 50 Appendix G limits being exceeded.

There are several means available for determining that there is not a temperature differential of > 50°F between the secondary and primary systems with $\leq$ 400 psig primary system pressure. These methods may include but are not necessarily limited to the following:

1) Converting steam line pressure indication into maximum temperature of steam generator fluid.

2) Tagging RCP switches with shutoff temperatures.

*Single failure considerations apply to active components.

3) Assuring adequate time for temperature gradients to dissipate.

4) Filling steam generators with water of known temperature.

REFERENCES:

(1) Final Engineering Report and Safety Analysis, Sections 9 and 10.

(2) Final Engineering Report and Safety Analysis, Paragraph 10.2.

(3) Supplement No. 1 to Final Engineering Report and Safety Analysis, Section 3, Question 9.

(4) NRC letter dated June 11, 1980, from D. G. Eisenhut to all operating pressurized water reactors.

(5) Letter to A. Schwencer from K. Baskin dated October 12, 1977.

3.1.3 COMBINED HEATUP, COOLDOWN AND PRESSURE LIMITATIONS

APPLICABILITY: Applies to heatup and cooldown of the reactor coolant system.

OBJECTIVE: To maintain the structural integrity of the reactor coolant system throughout the lifetime of the plant.

SPECIFICATION:

A. Reactor pressure and heatup and cooldown of the reactor coolant system during the first 16 years of equivalent full power operation shall be limited in accordance with Figures 3.1.3a and 3.1.3b. Thereafter, limits shall be based on neutron exposure equivalent to not less than 16 years of full power operation, and Figures 3.1.3a and 3.1.3b shall be updated accordingly (by formal license amendment application).*

B. Figures 3.1.3a and 3.1.3b shall be updated in accordance with the following criteria and procedures:

(1) The methods of Appendix G, "Protection Against Nonductile Failure", to Section III of the ASME Boiler and Pressure Vessel Code shall be used to obtain the allowable pressure-temperature relationships for the reactor coolant system.

(2) The curves in Figure 3.1.3c shall be used in predicting the reference nil-ductility temperature increase, RT_{NDT} unless measurements on the irradiation specimens show RT_{NDT}s greater than those predicted by the curves, in which case a new curve having the same slope as the original shall be constructed.

C. The pressurizer heatup rate of 100°F/hour and cooldown rate of 200°F/hour shall not be exceeded.

D. The reactor shall not be brought to a critical condition until the pressure-temperature state is to the right of the criticality limit line as shown in Figures 3.1.3a.

BASIS: The initial Reference Nil Ductility Temperature (RT_{NDT}) for all reactor vessel material based on Charpy V-notch data, drop weight tests, and conservative estimates** is 82°F or less. The RT_{NDT} at the 1/4 thickness location (location of Appendix G reference flaw tip) increases as a function of cumulative neutron exposure up to approximately 240°F for the core region of the reactor vessel after 30 years of operation.

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* Technical Specification 3.20.A(1) should be reevaluated for continued applicability of the low pressure PORV overpressure setpoint at any time the heatup and cooldown curves are changed.

** NRC Standard Review Plan Branch Technical Position MTEB 5-2.

A sixteen (16) equivalent full power year service period was chosen for the operational limits given in this specification because at the end of this period the limiting RT_{NDT} of the reactor vessel at the 1/4 thickness location is approximately 217°F in the core region. This RT_{NDT} is at least 50°F above the RT_{NDT} of all other regions in the primary reactor coolant system.

The highest RT_{NDT} of the core region material is determined by adding the radiation induced ΔRT_{NDT} for the applicable time period to the original RT_{NDT} shown in the Table 3.1.3.1. The fast neutron ($E > 1Mev$) fluence at 1/4 thickness and 3/4 thickness vessel locations is given as a function of full power service life in Figure 3.1.3d. Using the applicable fluence at the end of the year period and the copper content of the material in question, the RTNDT is obtained from Figure 3.1.3c.

Values of ΔRT_{NDT} may continue to be determined in this manner unless measurements on the irradiation specimens show ΔRT_{NDT}s greater than those predicted by the curves for the equivalent capsule exposure.

Allowable pressure-temperature relationships for various heatup and cooldown rates are calculated using methods derived from non-mandatory Appendix G in Section III of the ASME Boiler and Pressure Vessel Code, and discussed in detail in Reference 1.

The results of these calculations are provided in Reference 2.

The design heatup and cooldown rates for the pressurizer are 100°F/hour and 200°F/hour, respectively.

The vertical line portion of the criticality limit given in Figures 3.1.3a is at the minimum permissible temperature for the 2485 psig in-service hydrostatic test as required by Appendix G to 10CFR Part 50. The non-vertical portion of the criticality limit is shifted 40°F to the right of the heatup curve as required by Appendix G to 10CFR Part 50.

REFERENCES:

(1) "Pressure Temperature Limits" Section 5.3.2 of Standard Review Plan, NUREG-751087, 1975.

(2) S. E. Yanichko, et al, "Analysis of Capsule F from the Southern California Edison Company San Onofre Reactor Vessel Radiation Surveillance Program", WCAP 9520, May 1979.

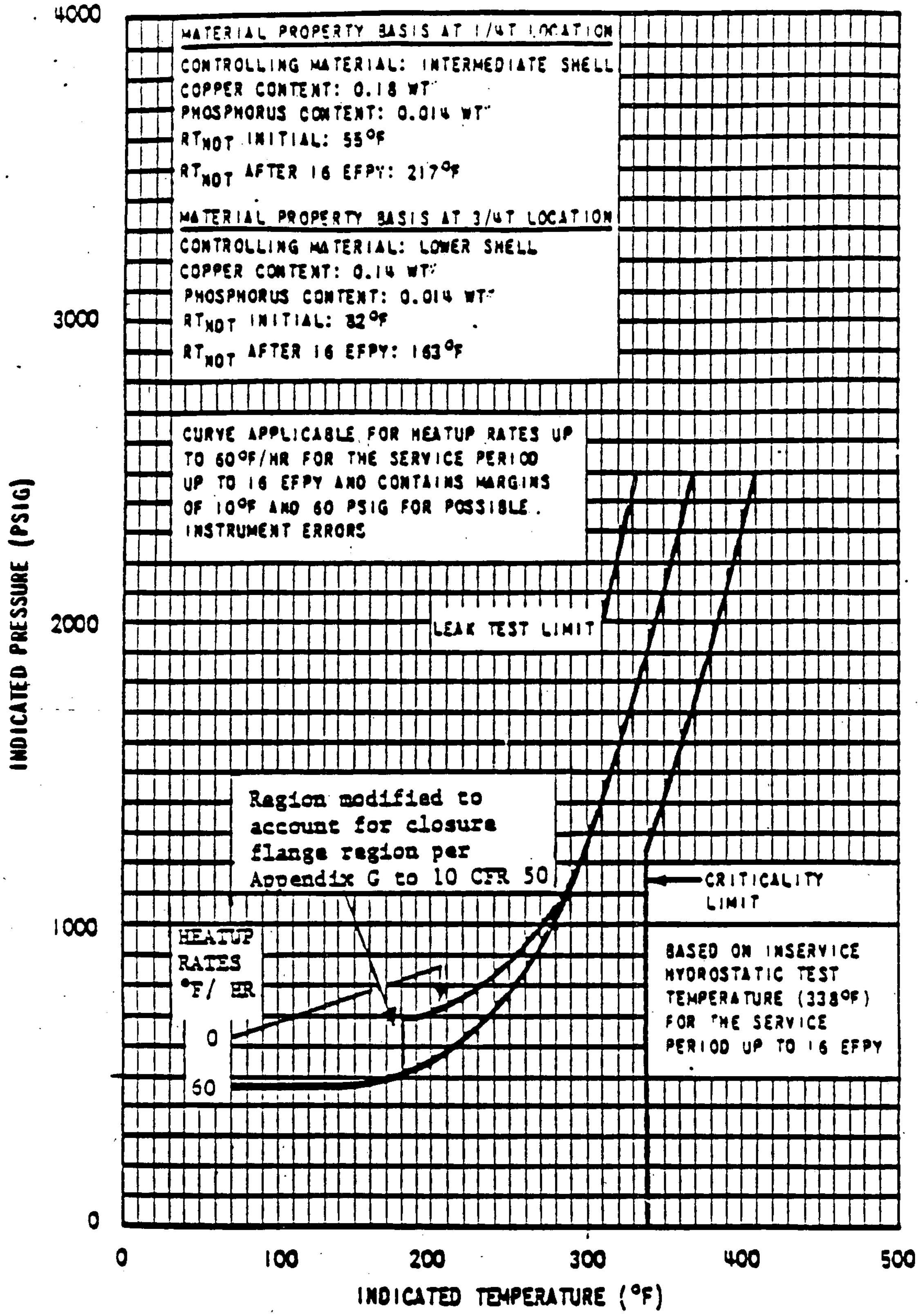

FIGURE 3.1.3a San Onofre Unit No. 1 Reactor Coolant System Heatup Limitations Applicable for the First 16 EFPY.

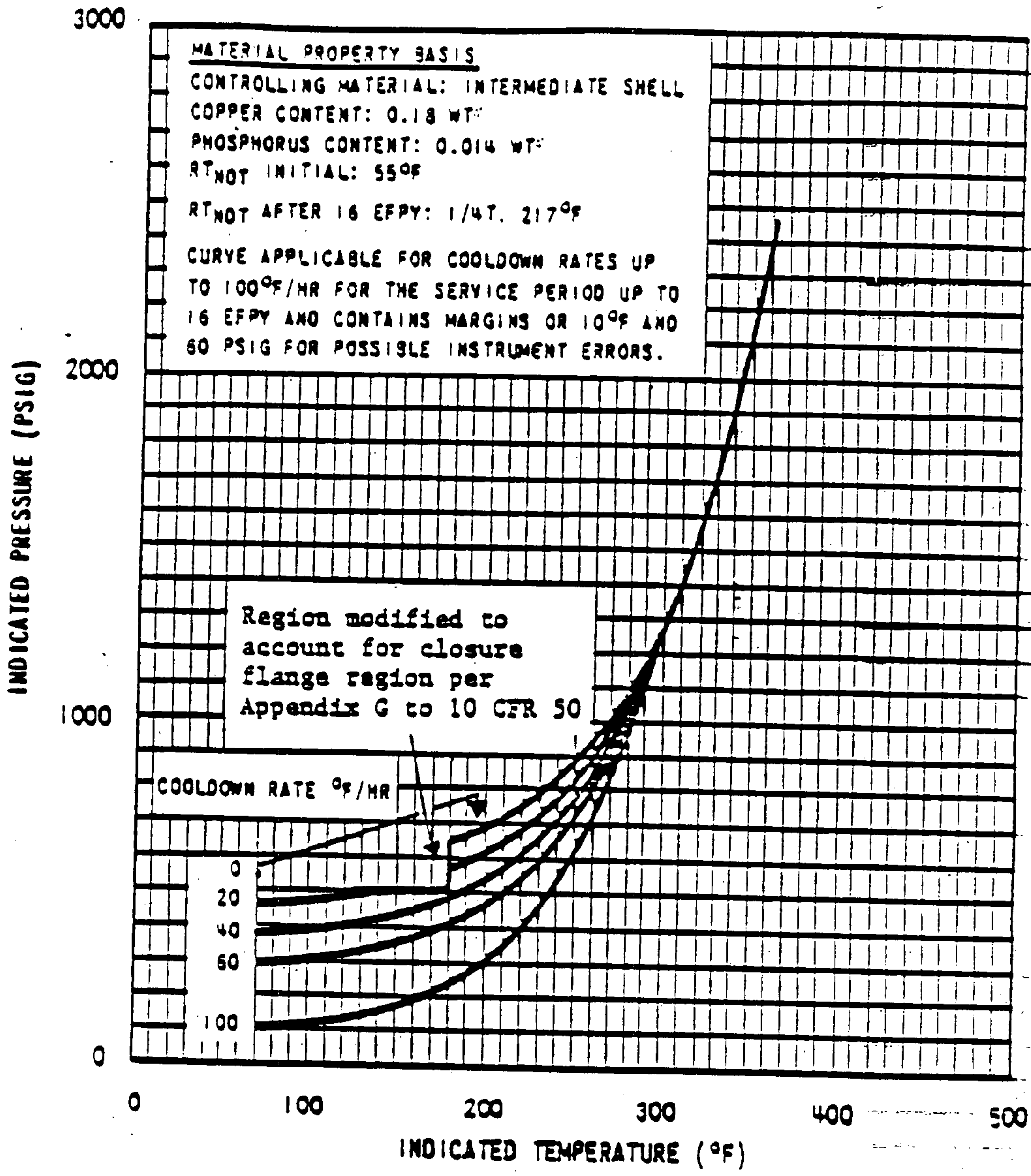

FIGURE 3.1.3b San Onofre Unit No. 1 Reactor Coolant System Cooldown Limitations Applicable for the First 16 EFPY.

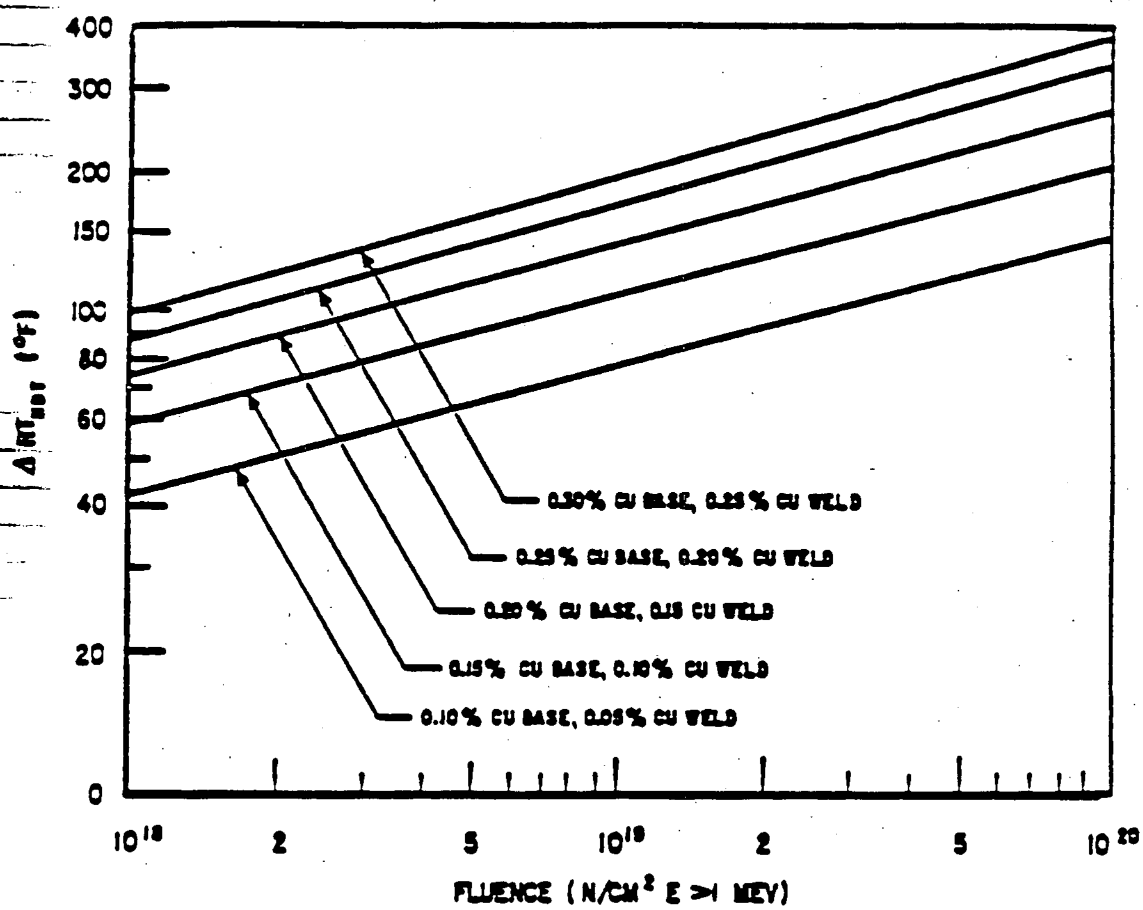

FIGURE 3.1.3c Effect of Fluence and Copper Content on ΔRT_{NDT} For Reactor Vessel Steels Exposed to Irradiation at 550°F

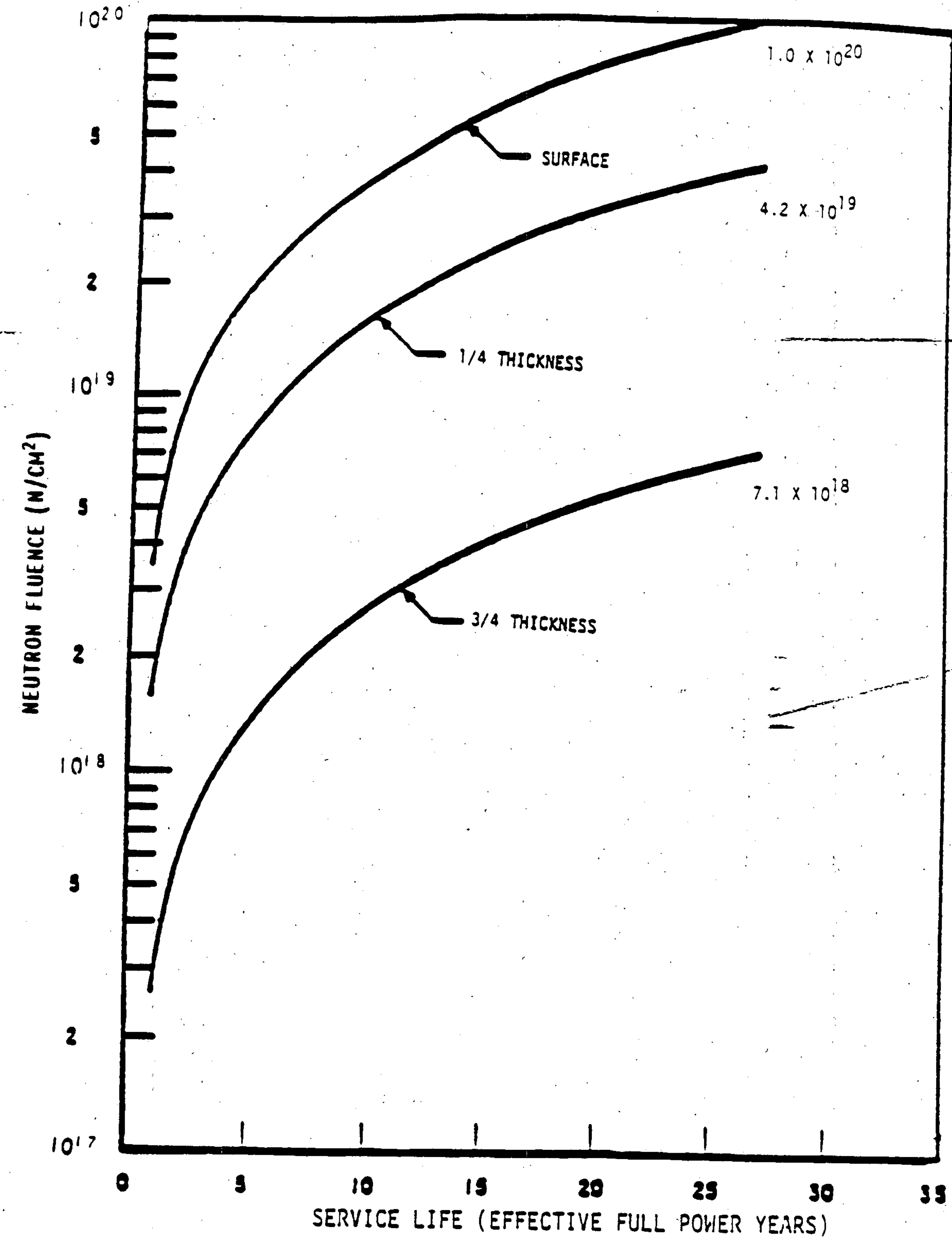

Figure 3.1.3d Fast Neutron Fluence (E > 1 MEV) as a Function of Full Power Service Life

TABLE 3.1.3.1

REACTOR VESSEL TOUGHNESS DATA (UNIRRADIATED)

Component	Code No.	Material Type	Cu (%)	P (%)	NDTT (°F)	Minimum 50 ft-lb/35 mil Temp (°F) Long.	Minimum 50 ft-lb/35 mil Temp (°F) Trans.	RT_{NDT} (°F)	Average Upper Shelf Energy (ft-lb) Long.	Average Upper Shelf Energy (ft-lb) Trans.
Cl. Hd. Dome	W7604	A302B	--	--	60(a)	112	132	72	72.5	--
Peel Segment	W7605-1	A302B	--	--	-10	114	134	74	70.5	--
Peel Segment	W7605-2	A302B	--	--	-10	90	110	50	122	--
Peel Segment	W7605-3	A302B	--	--	-10	108	128	68	85	--
Peel Segment	W7605-4	A302B	--	--	-10	120	140	80	74	--
Peel Segment	W7605-5	A302B	--	--	-10	26	46	10	109	--
Peel Segment	W7605-6	A302B	--	--	-10	102	122	62	88	--
Hd. Flange	W7602	A336 mod	--	--	60(a)	(b)	--	60	--	--
Ves. Flange	W7603	A336 mod	--	--	60(a)	(b)	--	60	--	--
Inlet Nozzle	W7611-1	A336 mod	--	--	60(a)	(b)	--	60	--	--
Inlet Nozzle	W7611-2	A336 mod	--	--	60(a)	(b)	--	60	--	--
Inlet Nozzle	W7611-3	A336 mod	--	--	60(a)	(b)	--	60	--	--
Outlet Nozzle	W7610-1	A336 mod	--	--	60(a)	(b)	--	60	--	--
Outlet Nozzle	W7610-2	A336 mod	--	--	60(a)	(b)	--	60	--	--
Outlet Nozzle	W7610-3	A336 mod	--	--	60(a)	(b)	--	60	--	--
Upper Shell	W7601-3	A302B	0.15	0.014	-10	48	68	8	98.5	--
Upper Shell	W7601-6	A302B	0.16	0.012	-30	64	84	24	104	--
Upper Shell	W7601-7	A302B	0.15	0.014	-20	52	72	12	95.5	--

a. Estimated per NRC Standard Review Plan Branch Technical Position MTEB 5-2.
b. Only 10°F Charpy V-notch data available. Conservative estimates for NDTT and RT_{NDT} were used.

TABLE 3.1.3.1(cont'd)

REACTOR VESSEL TOUGHNESS DATA (UNIRRADIATED)

Component	Code No.	Material Type	Cu (%)	P (%)	NDTT (°F)	Minimum 50 ft-lb/35 mil Temp (°F) Long.	Minimum 50 ft-lb/35 mil Temp (°F) Trans.	RT_{NDT} (°F)	Average Upper Shelf Energy (ft-lb) Long.	Average Upper Shelf Energy (ft-lb) Trans.
Inter. Shell	W7601-1	A302B	0.17	0.013	0	57	120(a)	60	94	75
Inter. Shell	W7601-8	A302B	0.18	0.012	10	93	100(a)	40	97	79
Inter. Shell	W7601-9	A302B	0.18	0.014	0	64	115(a)	55	102	72
Lower Shell	W7601-2	A302B	0.17	0.013	-20	74	94	34	97	--
Lower Shell	W7601-4	A302B	0.14	0.014	-10	91	111	51	94	--
Lower Shell	W7601-5	A302B	0.14	0.014	10	122	142	82	87.5	--
Bot. Hd. Peel	W7607	A302B	--	--	-20	62	82	22	91	--
Bot. Hd. Dome	W7606	A302B	--	--	60(b)	99	119	60	86	--
Weld	--	--	0.19	0.017	0(b)	--	29(a)	0	--	90
HAZ	--	--	--	--	0(b)	--	-14(a)	0	--	101

a. Actual not estimated.

b. Estimated per NRC Standard Review Plan Branch Technical Position MTEB 5-2.

3.1.5 PRESSURIZER RELIEF VALVES

APPLICABILITY: MODES 1, 2 and 3.

OBJECTIVE: To ensure reliability of the power operated relief valves (PORVs) and their associated block valves.

SPECIFICATION: Two PORVs and their associated block valves shall be OPERABLE.

ACTION:

A. With one or more PORV(s) inoperable, within 1 hour either restore the PORV(s) to OPERABLE status or close the associated block valve(s) and maintain the block valve(s) in the closed position; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

B. With one or more block valve(s) inoperable, within 1 hour restore the block valve(s) to OPERABLE status; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

C. The provisions of Specification 3.0.4 are not applicable.

BASIS: The power operated relief valves (PORVs) operate to relieve RCS pressure below the setting of the pressurizer code safety valves. These relief valves have remotely operated block valves to provide a positive shutoff capability should a relief valve become inoperable. The air supply for both the relief valves and the block valves is capable of being supplied from a backup passive nitrogen source to ensure the ability to seal this possible RCS leakage path.

REFERENCES:

(1) NRC letter dated July 2, 1980, from D. G. Eisenhut to all pressurized water reactor licensees.

3.2 CHEMICAL AND VOLUME CONTROL SYSTEM

APPLICABILITY: Applies to the operational status of the chemical and volume control system.

OBJECTIVE: To identify those conditions of the chemical and volume control system necessary to ensure safe reactor operation.

SPECIFICATION: A. When fuel is in the reactor, the following chemical and volume control system conditions shall be met:

(1) One charging pump or the test pump shall be OPERABLE. However, when the RCS pressure is < 400 psig and pressurizer water level is greater than 50%, a maximum of one of the two centrifugal charging pumps shall be OPERABLE. The inoperable centrifugal charging pump shall have the motor circuit breaker removed from the electrical power supply circuit and shall be condition tagged.

(2) One boric acid transfer pump or the boric acid injection pump shall be OPERABLE.

(3) A solution of at least 3450 pounds of boric acid in not less than 3500 gallons of water at a temperature of 140°F or higher, with at least one heater OPERABLE, shall be in the boric acid tank.

(4) System piping and valves shall be OPERABLE to the extent of establishing two flow paths for boric acid tanks.

(5) During periods when borated water is in the refueling cavity, the requirements in A.(1) through A.(4) may be waived provided that an alternate source of borated water is available to establish at least one flow path to the core for boric acid injection which can be initiated from the control room. The minimum capability for boric acid addition shall be equivalent to that supplied by a charging pump from the refueling water storage tank.

B. The reactor shall not be made critical unless the following additional conditions are met:

(1) One additional charging pump or test pump OPERABLE.

(2) One additional boric acid transfer pump or boric acid injection pump OPERABLE.

(3) Electrical heat tracing for boric acid piping OPERABLE.

C. After criticality is achieved, maintenance on item B.(3) will be allowed providing the boric acid temperature does not fall below 140°F.

BASIS: The Chemical and Volume Control System[1] provides control of the reactor system boron concentration. This is accomplished by using either one of the two charging pumps or the test pump (Chemical and Volume Control System test pump installed in parallel with the charging pump) to inject concentrated boric acid solution into the reactor coolant system. There are two sources of borated water available for injection through three different paths as follows:

1. The boric acid injection pump can deliver the boric acid tank contents to the charging pump and/or test pump.

2. Boric acid transfer pumps can deliver the boric acid tank contents to the charging pumps and/or test pump.

3. The charging pumps and the test pump can take suction directly from the refueling water storage tank (3750 ppm boron).

The quantity of boric acid in storage from the above two sources is sufficient to borate the reactor coolant in order to reach cold shutdown at any time during the core life. Furthermore, if the letdown capability from the primary coolant system to the chemical and volume control system should be impaired, the pressurizer void space volume is sufficient to accommodate the required injection. This free volume will accommodate sufficient concentrated boric acid solution such that the reactor coolant water can reach a concentration of about 400 ppm above the required level to shut the plant down.

The above system assured that for Specification A, continuous borated water supply is provided to maintain the core subcritical. In Specification B, redundancy is provided for borated water injection during reactor operations.

The limitation for a maximum of one centrifugal charging pump to be OPERABLE with an RCS pressure $\leq$ 400 psig with pressurizer water level greater than 50%, provides assurance that a mass addition pressure transient can be relieved by operation of the overpressurization mitigating system assuming a single failure of one PORV and no operator action for 10 minutes.

Tagged, as it applies to the inoperable charging pump, means tagged in accordance with current Southern California Edison procedures for tagging of equipment which must not be operated.

REFERENCE: (1) Final Engineering Report and Safety Analysis, Paragraph 3.6.

3.20 OVERPRESSURE PROTECTION SYSTEMS

APPLICABILITY: Applies to operability of the overpressurization protection systems.

OBJECTIVE: To preclude the potential for exceeding 10 CFR 50, Appendix G, in the event of a ~~pressure~~ transient while water-solid.

SPECIFICATION: A. When the RCS pressure is ≤ 400 psig* and pressurizer water level is greater than 50%, at least one of the following overpressure protection systems shall be OPERABLE:

(1) Two power operated relief valves (PORVs) with a lift setting of ≤ 500 psig,** or

(2) A reactor coolant system vent(s) of ≥ 1.75 square inches.

ACTION: B. With one PORV inoperable when required in accordance with Specification A above, either restore the inoperable PORV to OPERABLE status within seven days or depressurize and vent the RCS through a 1.75 square inch vent(s) within the next eight hours; maintain the RCS in a vented and tagged condition until both PORVs have been restored to OPERABLE status.

C. With both PORVs inoperable when required in accordance with Specification A above, depressurize and vent the RCS through at least a 1.75 square inch vent(s) within eight hours; maintain the RCS in a vented and tagged condition until both PORVs have been restored to OPERABLE status.

D. In the event either the PORVs or the RCS vent(s) are used to mitigate a potential RCS pressure transient, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 30 days. The report shall describe the circumstances indicating transient, the effect of the PORVs or vent(s) on the transient and any corrective action necessary to prevent recurrence.

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* The placing in service of the OMS at ≤ 400 psig is intended to assure that protection is provided whenever temperature is below 360°F. The alarm to arm the OMS being keyed to pressure assures that inadvertent opening of the PORVs does not occur due to placing the OMS into service with RCS pressure above the 500 psig initiation setpoint.

** The 500 psig setpoint is based on the current heatup and cooldown curves for 16 EFPY. The setpoint requires reevaluation for acceptability any time the curves are changed.

BASIS: The OPERABILITY of two PORVs or an RCS vent opening of greater than 1.75 square inches ensures that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when the initial RCS pressure is ≤ 400 psig and the pressurizer water level is greater than 50%, assuming a single failure of one PORV and no operator action for 10 minutes. Either PORV has adequate relieving capability to protect the RCS from overpressurization due to a design basis transient as described in submittal to the NRC dated October 12, 1977.

Tagged as it refers to the RCS vent, means tagged in accordance with current Southern California Edison procedures for tagging of equipment which must not be operated.

4.1.6 PRESSURIZER RELIEF VALVES

APPLICABILITY: Applies to the power operated relief valves (PORVs) an their associated block valves for MODES 1, 2 and 3.

OBJECTIVE: To ensure the reliability of the PORVs and block valves.

SPECIFICATION:

A. Each PORV shall be demonstrated OPERABLE:

1. At least once per 31 days by performance of a CHANNEL TEST, which may include valve operation, and

2. At least once per 18 months by performance of a CHANNEL CALIBRATION.

B. Each block valve shall be demonstrated OPERABLE at least once per 92 days by operating the valve through one complete cycle of full travel, unless the block valve is being maintained closed in order to meet the requirements of Specification 3.1.5.A.

C. The backup nitrogen supply for the PORVs and block valves shall be demonstrated OPERABLE at least once per 18 months by transferring motive power from the normal air supply to the nitrogen supply and operating the valves through a complete cycle of full travel.

BASIS: The power operated relief valves (PORVs) operate to relieve RCS pressure below the getting of the pressurizer code safety valves. These relief valves have remotely operated block valves to provide a positive shutoff capability should a relief valve become inoperable. The air supply for both the relief valves and the block valves is capable of being supplied from a backup passive nitrogen source to ensure the ability to seal this possible RCS leakage path.

REFERENCES:

(1) NRC letter dated July 2, 1980, from D. G. Eisenhut to all pressurized water reactor licensees.

4.20 OVERPRESSURE PROTECTION SYSTEMS

APPLICABILITY: Applies to OPERABILITY of the overpressurization protection systems.

OBJECTIVE: To verify that the overpressure protection systems will respond promptly and properly if required.

SPECIFICATION: A. Each power operated relief valve (PORV) shall be demonstrated operable by:

(1) Adjusting the pressure control bistable setpoint such that the PORVs are actuated and the annunciators alarm within 31 days prior to returning to a water-solid condition following a COLD SHUTDOWN with the RCS depressurized.

(2) Performance of a CHANNEL TEST within 31 days prior to enabling the low pressure overpressure mitigation setting of the pressurizer PORVs on cooldown.

(3) Performance of a CHANNEL CALIBRATION on the PORV actuation channel at least once per 18 months.

(4) Verifying that position indications on the PORV isolation valves indicate that the valves are open at least once per week when the PORVs are being used for overpressure protection.

BASIS: The surveillance requirement to verify OPERABILITY of the PORVs provides assurance that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when the initial RCS pressure is $\leq$ 400 psig. Either PORV has adequate relieving capability to protect the RCS from overpressurization due to a design basis transient as discussed in Reference 1.

REFERENCE:

(1) Letter to A. Schwencer from K. Baskin dated October 12, 1977.