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February 2, 1988
Fort St. Vrain
Unit No. 1
P-88045

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555

Attention: Mr. Jose A. Calvo
Director, Project Directorate IV

Docket No. 50-267

SUBJECT: Technical Specification Upgrade Program (TSUP) Additional Information

REFERENCES: See Attachment 1

Dear Mr. Calvo:

This letter provides the additional information requested in Enclosure 2 to Reference 1, to support NRC review of the Technical Specification Upgrade Program (TSUP). This submittal reflects the discussions and agreements reached during meetings held between Public Service Company of Colorado (PSC) and the NRC on August 25 and 26, 1987, as documented in Reference 2. All other information requested in Reference 1 was previously provided in Reference 3.

Attached to this letter are the following:

Attachment 2 provides a discussion for each NRC comment provided in Enclosure 2 to Reference 1. The NRC comment is repeated or summarized, PSC's responses and the results of the August 25-26, 1987 discussions are provided, and the final resolution category is identified.

A001
1/1

Attachment 3 provides proposed revisions for each Draft Upgraded Technical Specification that PSC agreed to change, as indicated in Attachment 2.

Attachment 4 provides a response to a previous NRC question about the need for a Technical Specification to limit the PCRV Support Ring maximum temperature. This comment was PSC Action Item 2 in Enclosure 3 to Reference 3.

In response to a question in Reference 2, PSC's current plans for the Fire Protection Technical Specifications are to delete them from the TSUP and implement the requirements via the Fire Protection Program Plan. Subsequent to the NRC's approval of the Fire Protection Program Plan, PSC will submit an amendment request to delete LCO 4.10 and SR 5.10 from the current Technical Specifications. PSC intends to also delete the following from the TSUP:

3/4.7.6	Fire Suppression Systems
3/4.7.6.1	Spray and/or Sprinkler Systems
3/4.7.6.2	Carbon Dioxide Systems
3/4.7.6.3	Halon Systems
3/4.7.6.4	Fire Hose Stations
3/4.7.6.5	Yard Fire Hydrants and Hydrant Hose Houses
3/4.7.7	Fire Rated Barriers

Consistent with the guidance previously provided by the NRC, PSC will retain the Fire Protection Technical Specifications in the TSUP until the NRC approves their removal.

PSC is in the process of preparing a Final Draft of the TSUP, to be submitted by April 11, 1988, per Reference 4. The remaining tasks that PSC will complete either prior to that submittal or in conjunction with that submittal are as follows:

1. Provide an Interlock Sequence Switch (ISS) report describing the different ISS positions and the different Reactor Mode Switch positions that are required to test various plant protective system circuits, by February 26, 1988.

2. Submit a QA Plan describing the processes PSC is utilizing to ensure the TSUP accurately reflects the FSV licensing basis, by February 26, 1988.

3. Revise TSUP sections to include the Technical Specification amendments that have been received since the TSUP began, by February 26, 1988.

4. Complete all open items remaining from the Reference 5 submittal, by February 26, 1988.

5. Prepare justifications for all "C" category comments, for submittal with the April 11 Draft Specifications.

The April 11, 1988 TSUP Draft will be reviewed by the FSV Plant Operating Review Committee (PORC) as stated in Reference 4. The redrafted Technical Specifications in Attachment 3 represent PSC's current positions on the identified issues. It is anticipated, however, that some further changes will be required during the PORC review.

If you have any questions regarding this information, please contact Mr. M. H. Holmes at (303) 480-6960.

Very truly yours,

Lawrence Brey

H. L. Brey
Manager, Nuclear
Licensing and Fuels

HLB/SWC/lmb

Attachments

cc: Regional Administrator, Region IV
ATTN: Mr. T. F. Westerman, Chief
Projects Section B

Mr. Robert Farrell
Senior Resident Inspector
Fort St. Vrain

P-88045
Attachment 1

REFERENCES: 1) NRC letter, Heitner to Williams, dated 7/2/87 (G-87217)

2) NRC memorandum, Heitner to Calvo, dated 10/1/87 (G-87348)

3) PSC letter, Brey to Calvo, dated 12/23/87 (P-87441)

4) NRC letter, Hunter to Lee, dated 8/22/85 (G-85354)

5) NRC memorandum, Heitner to Calvo, dated 1/12/88 (G-88009)

RESPONSE TO NRC

REQUEST FOR ADDITIONAL INFORMATION

PROVIDED IN ENCLOSURE 2

TO G-87217

RESPONSE TO NRC COMMENTS

This attachment addresses comments provided in Enclosure 2 to NRC letter dated July 2, 1987 (G-87217). For the most part, the NRC comments have been repeated in their entirety. A few of the more lengthy comments have been summarized.

Comment categories used in this Attachment are as follows:

- A PSC accepts comment as proposed
- A# PSC accepts comment with some changes or provides new wording
- B NRC accepts PSC position
- C NRC accepts PSC position and PSC will justify
- D PSC will review further
- D* NRC will review further
- F Comment is beyond scope of TSUP

The Resolutions documented herein reflect discussions and agreements reached during meetings between PSC and the NRC on August 25-27, 1987.

NRC Comment: RAI Table 1.0-1

The Licensee should provide additional justification for their proposed markup of Table 1.1 (P. 1-9, Attachment 1 of PSC letter of February 20, 1987) or should make the "@" and "#" footnotes specific to the precise evolutions (tests) or surveillances involved. PSC references such provisions in the GE-BWR STS. However, neither the GE-BWR STS, NUREG-0123, Rev. 3, nor the Perry TS (example given out in October 27-30, 1986 meeting by PSC) allow indiscriminate switching as would be allowed with the PSC proposed words of "...for the purpose of performing surveillances or other tests...". Both the GE-BWR STS and the Perry TS allow switching of the Mode Switch Position only under rather precisely defined conditions, such as "...to test the switch interlock functions..." or "...while a single control rod drive is being removed from the reactor pressure vessel per Specification 3.9.10.1..." or "...while a single control rod is being recoupled or withdrawn provided that the one-rod-out interlock is OPERABLE".

Further, please revise the pound sign (#) footnote in Table 1.1 by deleting the words "provided the keff is verified less than 0.99" and replacing the deleted words with "provided that the control rods are verified to remain fully inserted." As acknowledged in the PSC response to NRC Comment No. 4 on TSUP draft LCO 3.1.3, the use of the word "verified" is incorrect in establishing a quantitative estimate of SHUTDOWN MARGIN (that is, keff) because the reactor operators cannot actually verify the accuracy of the calculated assessments provided to them. However, consistent with the equivalent footnote provided in Table 1.2 of the BWR-STS (NUREG-0123, Rev. 3), the operators can verify that control rods are fully inserted.

PSC Response:

The first comment addresses the ISS setting footnote "@", where PSC had proposed that the ISS setting may be changed to the next higher mode setting, for the purpose of performing surveillances or other tests, for up to 72 hours, without being considered a change in operational modes. PSC is of the understanding that this is conceptually the type of control provided in BWR Tech Specs. Although the proposed words may lack the specificity provided in other Tech Specs, this has been compensated for by two additional features: 1) Switch settings are only permitted in one generally conservative direction, and 2) a 72 hour time restriction is imposed. PSC maintains that a comparable degree of control has been provided. The NRC accepted this position.

The second comment addresses the RMS setting, where PSC had proposed that the RMS setting may be changed for the purpose of performing surveillances or other tests, provided that keff is verified less than 0.99 by a second licensed operator or other qualified member of the unit technical staff. PSC agreed to the suggested "provided the control rods are verified to remain fully inserted," except for

Refueling operations or CRD testing per SR 4.1.1, which would require withdrawing control rods.

In response to NRC concerns about the lack of specificity in the application of the footnotes, PSC agreed to identify the testing that requires RMS or ISS setting changes. Also, PSC agreed to provide a report to clarify what interlocks are enabled by different ISS or RMS positions. This report will clarify why different switch positions are required to test different plant protective system circuits, and will be provided by February 26, 1988. The particular area of NRC interest involves changes for startup and low power operation.

Resolution: B, A#

NRC Comment: RAI SL 2.1.1 -5

The Licensee should provide additional information and a safety evaluation to support splitting the existing FSV Safety Limit 3.1, part into the proposed Safety Limit Section 2.1.1 (November 1985 Draft) and part into a Limiting Condition for Operation LCO 3.2.6 (November 1985 Draft). The 24 hr action time has been adequately addressed in PSC's letter of February 20, 1987. However, the reference to FSAR Revision 4 Section 3.6.8 does not provide specific justification for downgrading part of the existing Safety Limit to a Limiting Condition for Operation. In fact, FSAR Revision 4 Section 3.6.8 titled "Core Safety Limit" discusses all of the limits in the proposed SL 2.1.1 and LCO 3.2.6 as if they were safety limits as does the existing FSV SL 3.1.

PSC Response:

PSC does not believe that the existing Reactor Core Safety limit has been downgraded. Current SL 3.1 is admittedly confusing and the TSUP version is a significant clarification that is faithful to the original intent, as reviewed by GA.

Current SL 3.1 states that the combination of reactor core power-to-flow ratio and the total integrated operating time at the power-to-flow ratio during the lifetime of any segment shall not exceed certain limits (e.g., Fig. 3.1-1). It goes on to define what transients are to be evaluated against these limits by providing screening criteria (e.g., Fig. 3.1-2). The TSUP has retained the same Safety Limit as SL 2.1.1. The same screening criteria have been retained as LCO 3.2.6. This approach is consistent with the guidance provided by the STS wherein Safety Limit violations are preceded by LCO violations.

The NRC accepted PSC's justification and logic for splitting the existing SL 3.1 into LCO 3.2.6 and SL 2.1.1. This change will be documented in the formal justification that will be provided with PSC's amendment application.

Resolution: C

NRC Comment: RAI 3.0

Please provide an equivalent LCO to the recently proposed LCO 4.0.4 in the existing Technical Specifications.

PSC Response:

This comment addresses the guidance regarding use of the Calculated Bulk Core Temperature. PSC agreed to include the identical information as new Specification 3.0.5.

Resolution: A

NRC Comment: RAI 3.1.1 -1

Please revise the wording of TSUP draft LCO 3.1.1.a to add the words "from the fully withdrawn position" immediately after the words "152 seconds." PSC was to propose an alternative to the May 30, 1986, NRC markup of the TSUP draft. The proposed revisions to the basis does not adequately clarify the limiting condition for operation. In addition, the wording of the proposed insert at the bottom of page 3/4 1-5 in the PSC markup (Attachment 1 to P-87063) needs to be revised to read as follows:

> The full insertion scram time can be determined either directly from a full insertion scram time or indirectly from a partial scram time of 10 inches or more. For the partial scram time, the estimate of an extrapolated scram time is always based on assuming scram from the fully withdrawn position and not from the actual rod position.

PSC Response:

PSC agreed to the suggested changes as they accurately reflect the intent and manner in which this test is currently performed.

Resolution: A

NRC Comment: RAI 3.1.3 -4, -8

Please provide additional information with regard to the PSC position on the substantiation and verification of nuclear methods as expressed in the PSC response to NRC Comment Nos. 4 and 8 that are given in Attachment 2 to P-87063.

The NRC comment continues with several detailed questions.

PSC Response:

All FSV Shutdown Margin calculations for Tech Spec compliance are performed in accordance with written, approved procedures. They are independently verified and approved.

The NRC and PSC agreed, consistent with PSC's letter of August 15, 1986 (P-86496), that any additional questions regarding the justification of the reactor physics program at FSV should be addressed via separate, specific correspondence as they are beyond the scope of the TSUP.

Resolution: F

NRC Comment: RAI 3.1.6 -6

The Licensee should add SHUTDOWN and REFUELING to the Applicability statement of LCO 3.1.6 on the Reserve Shutdown System. Such added applicability should account for the exception required for any two control rod pairs which may be removed from the PCRV (LCO 3.1.4.2.a.1 of November 30, 1985 Draft). Although the May 30, 1986 NRC comment LCO 3.9.1-8 on inadequate SHUTDOWN MARGIN during REFUELING was categorized as an "F" (further discussion possible) in the October 1-2, 1986 meeting, further comparison to the GE-BWR STS Rev. 3, NUREG-0123, indicate that acceptance of small SHUTDOWN MARGINS (LCO 3.1.1 P 3/4 1-1) during REFUELING with any control rod withdrawn is with the provisio that the Standby Liquid Control System be OPERABLE (LCO 3.1.5, P 3/4 1-19). The GE-BWR STS is used for comparison rather than the W-STS as the Fort St. Vrain reactivity control system of control rods and reserve shutdown material is similar to the GE reactivity control system of control rods and Standby Liquid Control System, with neither having routine boration reactivity control as in the Westinghouse system. Also, for Fort St. Vrain, the Actions b.2 for SHUTDOWN and C.2.b for REFUELING with inadequate SHUTDOWN MARGIN (LCO 3.1.3 of November 30, 1985 Draft) require actuation of sufficient reserve shutdown material to achieve the required SHUTDOWN MARGIN. Yet in the Reserve Shutdown System Specification (LCO 3.1.6 of the November 30, 1985 draft) there is no requirement for OPERABILITY in either the SHUTDOWN or REFUELING condition.

PSC Response:

In the August 25-26, 1987 meeting, this item was left for further review and was categorized as a D comment. PSC has reviewed this Specification further and we agree to include the current interim Specification 3/4.1.9 in the TSUP.

Resolution: A

NRC Comment: RAI 3.2.1 -1

Please expand the Basis to describe briefly what activity is involved in the "determination by evaluation." What reactor observables are evaluated?

PSC Response:

PSC agreed to revise the Basis by replacing the last paragraph with the following:

> "The core irradiation limit of 1800 EFPDs is related to a residence time in the core for each element. An evaluation of the residence time records and an allowance for the duration of the next fuel cycle will ensure that this Specification is not exceeded during the next cycle of operation for any element."
>
> Resolution: A

NRC Comment: RAI 3.3.2.3 -3

The Licensee should add a calibration requirement in SR 4.3 2.3.2 for seismic instruments determined to be out of calibration following a seismic event. As indicated in the Basis, P. 3/4 3-83 of the November 30, 1985 Draft, the Licensee has already committed to this (last sentence, fourth paragraph). However, there is presently no requirement for such calibration in the surveillances. Although the STS guidance is to do such calibration within ten days, it is judged that 30 days provides for allowing monitoring of after shocks following the initial event. Experience with other commercial plants is that the contractor/manufacturer provides on-site calibration capability. PSC should investigate such on-site service capability from their seismic instrument manufacturer so that the seismic instruments remain on-site to be maximumly available for monitoring after shocks.

PSC Response:

PSC agreed to add a SR to calibrate seismic instruments that are found out of calibration following a seismic event, within 30 days.

Resolution: A

NRC Comment: RAI 3.3.2.3 -4

The Licensee should specify monthly CHANNEL CHECKS for the seismic instruments in TS Table 4.3.2-2 (November 1985 Draft, P. 3/4 3-82) rather than the proposed quarterly checks. PSC had proposed quarterly CHANNEL CHECKS as consistent with Turkey Point seismic instrument surveillances. However, as Turkey Point specifications are the exception to the rule and as Turkey Point is situated in Seismic Zone 0 and FSV is situated in Seismic Zone 1, the Turkey Point specifications are not a reasonable comparable to use. Additionally, it is in the Licensee's interest to have the seismic instruments operable to facilitate restart if shutdown occurred. The relatively uncomplicated CHANNEL CHECKS would enhance instrument operability and thus the assessment of a seismic disturbance and the implications to plant restart capability.

PSC Response:

In the case of the accelerographs, PSC notes that a Channel Check consists of a battery check. There is no instrument drift to examine, only mechanical movement. In PSC's experience, we have not observed problems with the batteries to the extent that a monthly check should be required. The potential for inadvertent actuation during bumping (the battery check requires a disconnection and re hook-up) and the resultant loss of the instrument availability, would indicate that less frequent handling is desirable.

In the case of the seismoscopes, it was agreed that a Channel Check would consist of shining a flashlight through a viewing window to see if the smoked glass has been etched due to an actuation of the device. For the readily accessible seismoscopes, this would be performed monthly. For the less accessible devices, a quarterly channel check is acceptable.

Resolution: Originally D*, updated to A# per telecon with NRC on January 11, 1988.

NRC Comment: RAI 3.5.4

Please revise LCO 3.5.4 ACTION b (second part on page 3/4.5-30 of the TSUP draft). The words, "establish a backup system for fire suppression purposes within 24 hours", need to be changed to the wording, "establish a backup system for fire suppression purposes prior to reaching a CALCULATED BULK CORE TEMPERATURE of 760 °F but within a period of time not to exceed 24 hours." The revision is necessary to assure an operable flow path to the liner cooling system (LCS) via the firewater system when the LCS is the only decay heat removal path during an interruption of forced cooling for purposes of maintenance and inspection. This change is consistent with the NRC guidelines provided in the NRC letter of December 5, 1986, on PSC commitments required to approve the proposed version to LCO 4.1.9 in the existing Technical Specifications.

PSC Response:

This section has been re-written as part of PSC's response to the RAI on safety related cooling system Tech Specs. The Specification has been re-designated as 3/4.5.5 and was submitted in P-87441, dated 12/23/87.

The Action to establish a backup system deals with fire suppression capabilities, which PSC has removed from Specification 3/4.5.5, in order to focus on the Safe Shutdown Cooling function of the firewater system.

The Actions for Specification 3/4.5.5 (submitted in P-87441) were agreed upon in a meeting with the NRC on December 2-3, 1987, as acceptable for assuring a flow path to the liner cooling system. No additional changes are required by this comment.

Resolution: B

NRC Comment: RAI 3.6.4 -2

Please provide additional information with regard to the source and approval of the acceptance criteria for PCRV concrete permeability and PCRV liner thinning as cited in the PSC markup of pages 3/4.6-39 and 3/4.6-40 of the TSUP draft in Attachment 1 to P-87063. The NRC does not have a copy of the previous ISI procedures alluded to in the citations, and these acceptance criteria are not found in the recent ISIT submittals.

PSC Response:

The acceptance criteria for PCRV concrete permeability and liner thinning have not been previously approved. The only basis is past practice, as identified in the Basis. These criteria were included in PSC's implementing procedures for these current Tech Specs (SR 5.2.13-X and SR 5.2.14-X, respectively). It was agreed that furthei justification beyond PSC's existing practice is beyond the scope of TSUP.

Resolution: F

NRC Comment: RAI 3.7.2 -4

The Licensee should provide additional clarification in the basis on P. 3/4 7-14, third paragraph, second and third sentences, as they read:

> "A 1 hour ACTION time is provided to isolate the affected loop, in an effort to regain OPERABILITY of a second hydraulic fluid pump and/or at least one accumulator for the affected valve group. If OPERABILITY of a second hydraulic fluid pump and/or at least one accumulator is not restored within 1 hour, reactor shutdown is required within 24 hours."

Since OPERABILITY of a second hydraulic fluid pump is not mentioned in the Action Statement, only "supply of at least 2500 psig", the basis should use the terminology of the Action Statement, namely, supply of at least 2500 psig. Also, the basis has interpreted the Action Statement to allow one hour to restore the required conditions of at least one accumulator and or at least 2500 psig pressure before reactor shutdown within the next 24 hours. The Action Statement a. of P. 3/4 7-12 does not address a restoration time. It requires isolation of the affected loop in one hour and reactor shutdown in the next 24 hours without regard to any restoration time.

PSC Response:

PSC agreed to revise the Basis statements as follows:

> "A 1 hour ACTION time is provided to isolate the affected loop, in an effort to regain the capability to supply at least 2500 psig and/or the OPERABILITY of at least one accumulator for the affected valve group. If these efforts are not successful, reactor shutdown is required within the next 24 hours."

Resolution: A

NRC Comment: RAI 3.7.6.3 -3

For the halon system in Building 10, please propose specifications consistent with SR 4.7.6.3 in the WNP No. 2 Technical Specifications, NUREG-1009 (see attached page). At WNP No. 2, tank "quantity" is determined once per six months using the heat tape and gun method approved by the American Nuclear Insurers. However, storage tank weight must be verified periodically. PSC needs to specify an appropriate surveillance period for verifying storage tank weight and to justify any period exceeding 36 months.

PSC Response:

PSC's Fire Protection Program Plan (PSC letter, Williams to Calvo, dated 12/15/87, P-87422) includes the operability requirements and periodic test requirements for the Halon Systems. This plan was developed through discussions with the NRC Fire Protection reviewers, separate from TSUP. PSC has revised Specification 3/4.7.6.3 so that the Building 10 Halon system surveillance is consistent with the Program Plan, and this includes a quantity check. Since PSC plans to move this Tech Spec into the Fire Protection Program Plan, per GL 86-10, PSC considers that further discussions regarding weight verification should best be addressed through the Fire Protection Program, separate form the TSUP.

Resolution: A# (originally A, but changed to A# to reflect the Fire Protection Program Plan positions)

NRC Comment: RAI 3.7.8 -4

Please correct the misspellings in PSC's markup of TSUP draft LCO 3.7.8 ACTION b. The word, "values," is misspelled twice as "valves."

PSC Response:

PSC agreed to make the corrections as noted. This Specification has been re-designated as 3/4.8.4 and is being submitted separately as part of the Electrical Tech Specs.

Resolution: A

NRC Comment: RAI 3.7.10 -3

Please revise FSAR Sections 1.4 and B.5.2.7 to be consistent with the Basis definition of "safety-related" as including Class 1a components. PSC expressed the desire to retain the wording in the opening sentence of the Basis as being indicative of their position with regard to the scope of the safety-related snubbers. The Basis and the FSAR need to be consistent on this point.

PSC Response:

PSC agreed to consider a revision to the FSAR, as requested, separate from the TSUP.

Resolution: F

NRC Comment: RAI 3.9.1 -2

The Licensee should retain the APPLICABILITY as it was in the November 1985 Draft or should provide additional information or changes to split out Specifications 3.9.1a and b from Specifications 3.9.1c and d. Specifications 3.9.1a and b are justly applicable to only "whenever both primary and secondary PCRV closures of any PCRV penetrations are removed" which is PSC's proposed APPLICABILITY. However, Specifications 3.9.1c and d on requiring two startup channel neutron flux monitors and maintaining the SHUTDOWN MARGIN requirements of Specification 3.1.3, are applicable throughout the REFUELING mode. Therefore, either the original APPLICABILITY in the November 1985 Draft should be retained or 3.9.1a and b need separated from 3.9.1c and d with different applicability statements as discussed above.

PSC Response:

In the November 1985 draft, Specification 3.9.1.c which requires two startup channel neutron flux monitors was only applicable during Core Alterations affecting Core Reactivity, per the ** note. These Core Alterations are only performed with both primary and secondary closures removed. The NRC considered that PSC should have two neutron flux monitors during the entire REFUELING mode and PSC agreed to propose a new Specification that would take these monitors out of Specification 3.9.1. The Action would be to suspend Core Alterations, similar to STS.

Specification 3.9.1.d is a cross-reference to Specification 3.1.3, Shutdown Margin, which is provided for emphasis. Specification 3.1.3 is applicable at all times, whether it is cross-referenced or not. The NRC accepted this position.

Resolution: A#, B

NRC Comment: RAI 3.9.1 -3

The Licensee should revise the wording as suggested for Action b of LCO 3.9.1 (similar wording should also be used for Action C.1 of this LCO) to:

> "With one of the above required neutron flux monitors inoperable, or not OPERATING, immediately suspend all operations involving CORE ALTERATIONS, any evolution resulting in positive reactivity changes, or movement of IRRADIATED FUEL."

The Licensee's position that the proposed words "...control rod movements resulting in positive reactivity changes..." were intended to be specific versus the NRC recommended wording ..."any evolutions resulting in positive reactivity changes..." has been accepted for the other LCOs involved... For the subject LCO, however, Actions b and C.1 involve loss of one and both startup channel neutron flux monitors, respectively. Allowing an intended positive reactivity change due to cooldown with degraded startup channel neutron flux monitoring is unacceptable. When both startup channels are inoperable, any controllable positive reactivity change should be stopped, as under these conditions, immediate assessment of flux changes is lost. With only one operable startup channel of neutron flux monitoring, it is also unacceptable to intentionally make positive reactivity changes as that one startup channel may be operating erroneously and there is no second operating channel to confirm its readings. Also, with only one operating channel, a sudden loss of that channel again results in complete loss of the ability to make immediate assessments of neutron flux changes. These changes would make the FSV Actions consistent with those of STS Rev. 5, P. 3/4 9-2, on neutron flux monitoring capability during refueling, and with those of the existing FSV Technical Specifications, LCO 4.7.1.

PSC Response:

PSC accepted the proposed changes for Actions b and C.1.

Resolution: A

NRC Comment: RAI 6.3/6.4 -1

The Licensee should include reference to the NRC March 28, 1980 letter in Technical Specification Sections 6.3.2 and 6.4.1 per the subject NRC comment. Contrary to the Licensee's position in the October 27-30, 1986 Meeting that the NRC March 28, 1980 letter is not in their letter log and therefore is doubtful that it was agreed to, the Licensee responded to it in their letter of December 20, 1980 (P-80438). In Attachment 1 of that letter, the Licensee stated that although they had received the NRC March 28, 1980 letter too late to implement the requirements by August 1, 1980 as required by the letter's Enclosure 1, some requirements would be met by the August 1, 1980 date and their program for compliance would be submitted by January 15, 1981. As it appears that the Licensee has committed to the subject NRC March 28, 1980 requirements, stating this in the Technical Specifications should not involve any undue hardships.

PSC Response:

The March 28, 1980 letter was included in NUREG 0737, Section I.A.2.1. PSC developed a training program to meet the intent of this letter, but in some areas, such as use of a simulator, PSC could not comply. PSC's Licensed Operator Requalification Program was approved by the NRC in their letter of June 4, 1987 (G-87185). PSC does not believe it would be appropriate to cite the March 28, 1980 letter in the Tech Specs because its requirements have not been met in their entirety. The NRC accepted this position.

Resolution: B

NRC Comment: RAI 6.5.1.6a

The Licensee should revise AC 6.5.1.6a to require the PORC to review any procedures required by AC 6.8.4 as AC 6.8.4 includes procedures for iodine sampling in the reactor building and Post-Accident Sampling, both of which are TMI-2 Action items of NUREG-0737 and Generic Letters 83-36, 37. Also, the Licensee should revise AC 6.5.2.7 to require the NFSC to review the programs of AC 6.8.4. This is what PSC said PORC and NFSC do now (See PSC letter of February 20, 1987, Attachment 2). However, as written, AC 6.5.1.6a and AC 6.5.2.7 do not have these requirements but should.

PSC Response:

AC 6.8.4 requires PSC to have Programs for In-Plant Radiation Monitoring, Secondary Water Chemistry, and Post-Accident Sampling. PSC's current Tech Specs do not require PORC or NFSC review of these programs. PORC currently does review implementing procedures for these programs and PSC agrees to add 6.8.4 to AC 6.5.1.6a.

Regarding NFSC review of these Programs, PSC intended that this fall under the category of AC 6.5.2.8.a, audits of facility operation to Tech Spec provisions. Specific direction to review these Program activities is inconsistent with the STS guidance provided by the NRC. The NRC accepted PSC's position that NFSC review of AC 6.8.4 program should not be explicitly addressed in the Technical Specifications.

Resolution: A, B

NRC Comment: RAI 6.5.1.7b

The Licensee should provide additional information to explain what "Procedure Deviation Reports" are, as used in their letter of February 20, 1987, Attachment 3, Item AC 6.5# 1-7 and Attachment 1, P. 6-15, marked up item 6.5.1.7.b. In Attachment 3, reference is made to "Temporary Changes" whereas in Attachment 2, reference is made to "Procedure Deviation Reports". It is not clear if these are one and the same thing. If they are the same, PSC should provide additional information to justify the proposed exception for "Temporary Changes".

PSC Response:

The "Procedure Deviation Reports" (PDR) used in the Tech Spec markup of AC 6.5.1.7.b is the same as the "temporary change" identified in PSC's discussion. The issue is that PORC does not approve PDRs prior to their implementation. A PDR is a change to a procedure that adds, deletes, or modifies activities in that procedure, as long as the procedure intent is not changed. Its use, including required approvals and goals for incorporation into a procedure revision, is controlled by Administrative Procedure G-2, "FSV Procedure Systems." The NRC accepted PSC's explanation of PDRs.

Resolution: B

NRC Comment: RAI 6.9.1.2.a

The Licensee should revise their proposed marked up P. 6-27 (PSC letter of February 20, 1987, Attachment 1) to either put report submittal times in 6.9.1.2, first paragraph as in the STS Rev. 5, P. 6-16, or place a report submittal directly in 6.9.1.2.a.2. As proposed, 6.9.1.2.a.2 does not have any report submittal time connected to it. PSC placed their report submittal directly in 6.9.1.2.a.1 rather than 6.9.1.2. Therefore, when 6.9.1.2.a.2 was added, it was not covered by a report submittal time.

PSC Response:

This comment deals with the Annual Occupations Exposure Report. PSC agreed to add a requirement to submit the specific activity analysis report of 6.9.1.2.a.2 to the NRC Regional Administrator by March 31 of each year.

Resolution: A#

NRC Comment: RAI NRC Action 3

PSC should provide the Environmental Qualification study for this item. The following is NRC Action Item 3 and its response from Attachment 1 to PSC letter of November 27, 1985 (P-85448).

NRC Action

Provide guidance on whether components, which are required to function to maintain other equipment within an environment for which it is qualified, should also be in the Technical Specifications (for example, main steam isolation valves).

NRC Response

Where assumptions for equipment operability related to environmental qualification are based on the successful operation of active components in the event of an accident, the availability and reliability of these components should be ensured through Technical Specification requirements. Specific items of concern identified by the staff were the main steam isolation valve (MSIV) operability and closing time requirements as well as the hot reheat (HRH) valve operability requirements. Other EQ operability requirements should be identified by the licensee and incorporated as Technical Specification requirements to ensure equipment necessary to mitigate accidents function within the assumed environmental conditions.

In addition to the EQ analysis, other analysis which rely on equipment operability may also result in Technical Specification requirements. For example, the equipment operability requirements based on previous fire protection analysis (Appendix R Evaluation: FSV Reports 1 through 4) should be reflected in the Technical Specifications.

PSC Response:

This comment requests PSC to provide a report identifying which active components must function to 1) preserve an environment for which equipment is qualified (e.g., SLRDIS actuated valves), and 2) safely shutdown the plant in the event of a fire.

PSC has provided an extensive discussion in the FSAR (Sections 1.4.5.3 and 7.3.10) regarding equipment which is required following a high energy line break (HELB) at FSV. The equipment which must function is identified, along with the conditions under which it must function and the assurances PSC has taken for proper functions. The NRC agreed that no additional report is required.

PSC agreed to consider a Technical Specification regarding the system valves that are actuated by the SLRDIS system to isolate a HELB. A draft of this specification will be provided in separate correspondence, prior to the submittal of the next draft of the TSUP.

It was also agreed that no new Technical Specifications are required for the Appendix R evaluation, consistent with the guidance provided in GL 86-10.

Resolution: A#

ATTACHMENT 3

TO P-88045

PROPOSED REVISIONS TO

DRAFT UPGRADED TECHNICAL SPECIFICATIONS

DISCUSSED IN ATTACHMENT 2

TO THIS LETTER

The Specifications included in this Attachment have been marked to show revisions from previous drafts, based on the discussions in Attachment 2.

(D - Deleted, R - Revised, N - New)

TABLE 1.1

OPERATIONAL MODES

MODE	INTERLOCK SEQUENCE SWITCH SETTING	REACTOR MODE SWITCH SETTING	% RATED THERMAL POWER*
POWER (P)	Power	Run	> 30%
LOW POWER (L)	Low Power @	Run	> 5% and ≤ 30%
STARTUP (S/U)	Startup @	Run	≤ 5%
SHUTDOWN (S/D)	**	Off #	0
REFUELING (R) ***	**	Fuel Loading #	0

* Excluding decay heat.

** Interlock Sequence Switch (ISS) may be in any position in SHUTDOWN and REFUELING.

*** Includes Reactor Internal Maintenance, See Specification 3/4.9.1.

\# The Reactor Mode Switch setting may be changed for the purpose of performing surveillances or other tests, provided the control rods are verified to remain fully inserted (or as otherwise required for Refueling operations or testing per SR 4.1.1) by a second licensed operator or other qualified member of the unit technical staff. R

@ The Interlock Sequence Switch setting may be changed to the next higher Mode setting, for the purpose of performing surveillances or other tests, for up to 72 hours, without being considered a change in OPERATIONAL MODES.

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LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

3.0.5 Where the Applicability of a FSV Technical Specification is defined in terms of the CALCULATED BULK CORE TEMPERATURE, the time at which this temperature reaches 760 degrees F following an interruption of all primary coolant flow is the time after which specification requirements are applicable. The time for the CALCULATED BULK CORE TEMPERATURE to reach 760 degrees F following an interruption of all primary coolant flow is determined as follows:

a. Using the applicable operating power history prior to interruption of primary coolant flow, determine the decay heat power from Figure 3.0-1.

b. Using this decay heat power and the average core temperature prior to the primary coolant flow interruption, determine the time required to reach 760 degrees F from Figure 3.0-2.

c. The maximum time for which primary coolant flow can be interrupted is the time interval determined in Specification 3.0.5.b, not to exceed 21 days.

N

4.0 SURVEILLANCE REQUIREMENTS

4.0.1 Surveillance Requirements shall be applicable only during the OPERATIONAL MODES or other conditions specified for individual Limiting Conditions for Operation unless otherwise stated in an individual Surveillance Requirement.

4.0.2 Each Surveillance Requirement shall be performed within the specified time interval with:

a. A maximum allowable extension not to exceed 25% of the SURVEILLANCE INTERVAL, and

b. The combined time interval for any 3 consecutive SURVEILLANCE INTERVALS not to exceed 3.25 times the specified SURVEILLANCE INTERVAL.

4.0.3 Failure to perform a Surveillance Requirement within the specified SURVEILLANCE INTERVAL shall constitute a failure to meet the OPERABILITY requirements for a Limiting Condition for Operation. Exceptions to these Surveillance Requirements are stated in the individual specifications. Surveillance Requirements do not have to be performed on inoperable equipment.

Fig. 3.0-1. FSV Decay Heat Power Fraction vs Time after Reactor Shutdown for Various Times at Constant Power

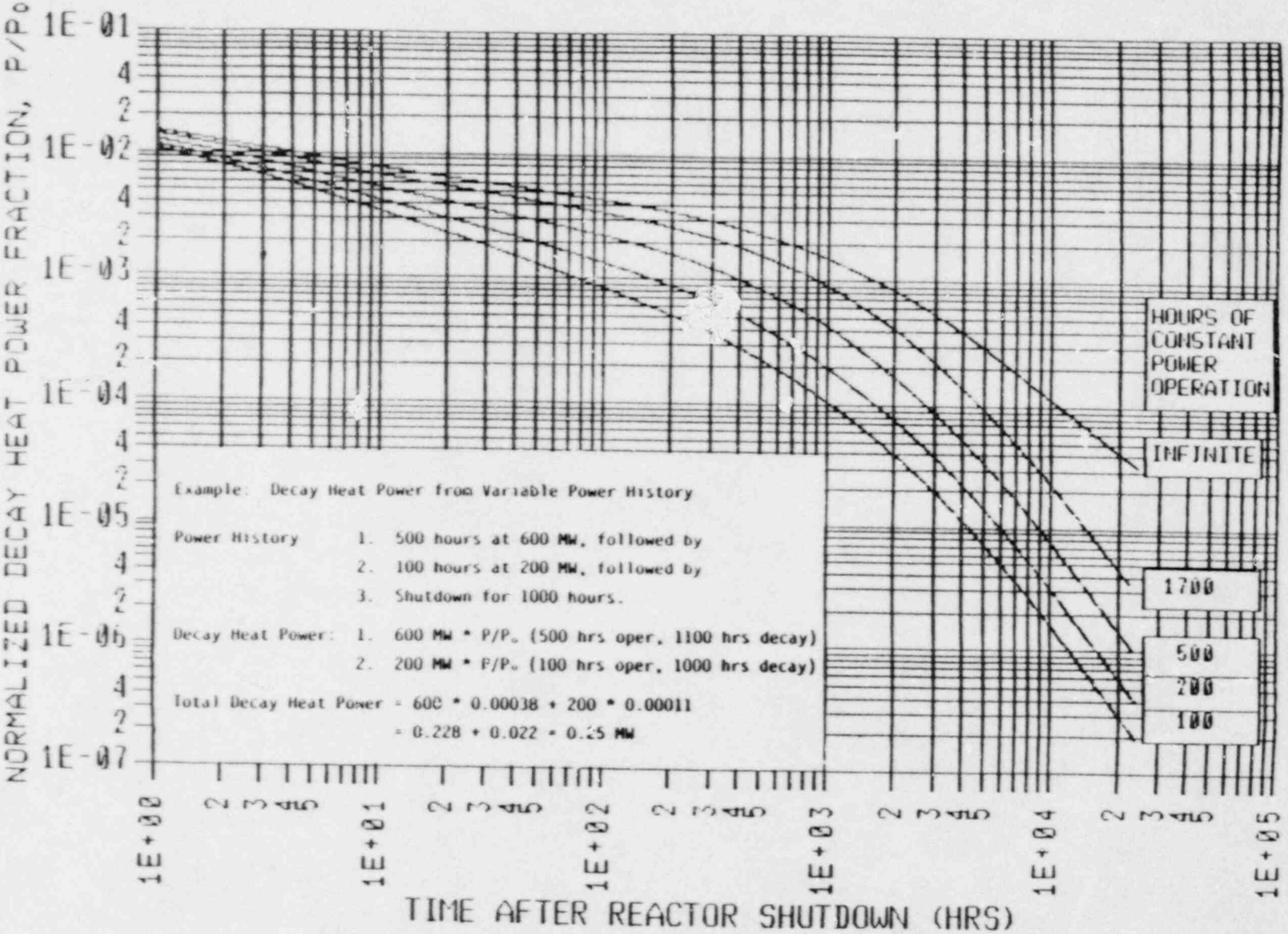

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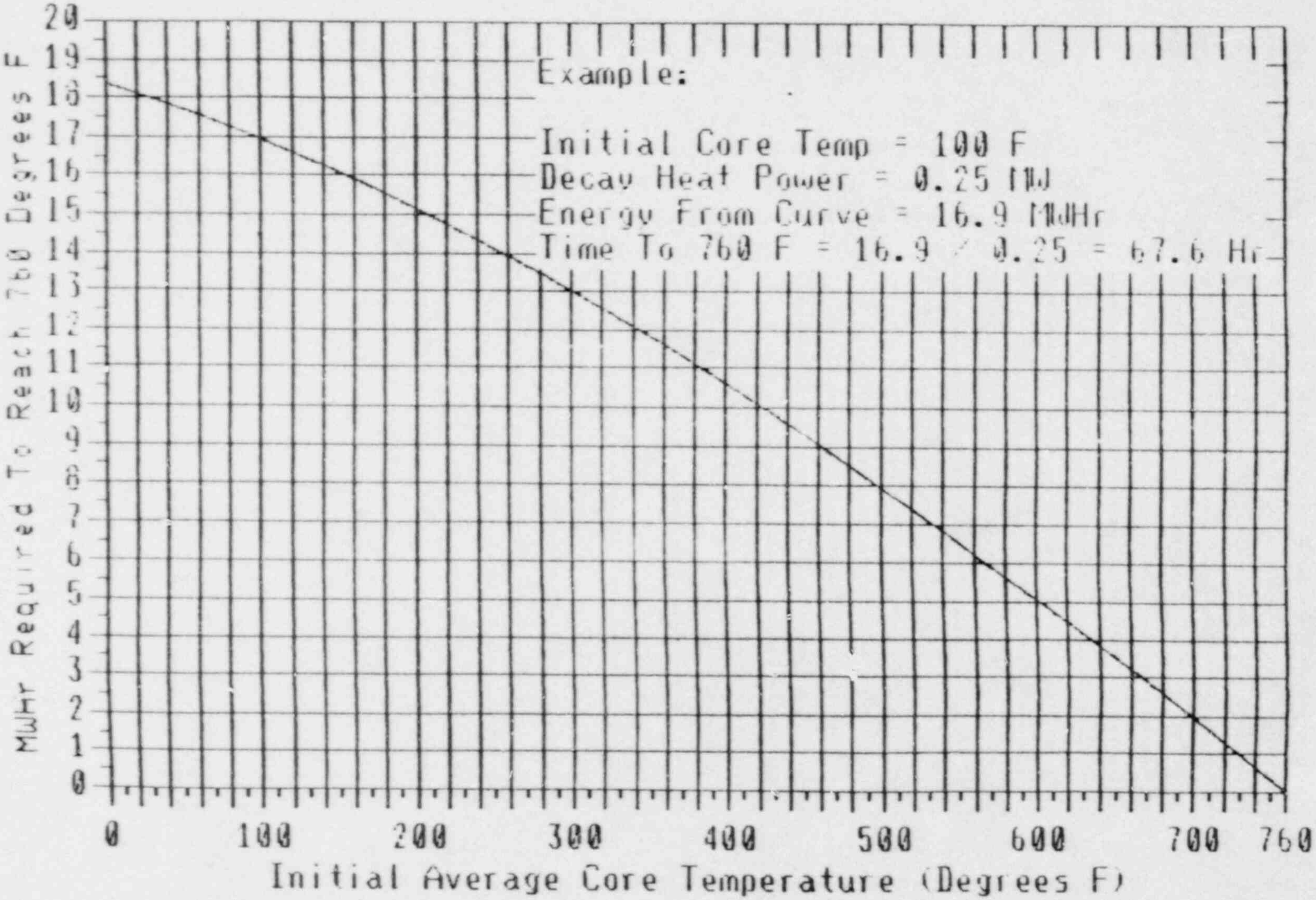
Fig. 3.0-2. Decay Heat Energy Required
To Raise The Core Temperature To 760 Degrees F



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BASIS FOR SPECIFICATION 3.0/SR4.0 (continued)

3.0.4 This Specification provides that entry into an OPERATIONAL MODE or other specified applicability conditions must be made with: (1) the full compliment of required systems, equipment, or components OPERABLE and (2) all other parameters as specified in the Limiting Condition for Operation being met without regard for allowable deviations and out-of-service provisions contained in the ACTION statements.

The intent of this provision is to ensure that facility operations is not initiated with either required equipment or systems inoperable or other specified limits being exceeded.

Exceptions to this provision have been provided for a limited number of specifications when startup with inoperable equipment would not affect plant safety. These exceptions are stated in the ACTION statements of the appropriate specifications.

3.0.5 The CALCULATED BULK CORE TEMPERATURE is the calculated, time dependent, average temperature of the core, including graphite and fuel, but not the reflector, assuming a loss of all forced circulation of primary coolant flow. The calculation uses several conservative assumptions: 1) The decay heat power at the start of the core heatup has been conservatively selected using Figure 3.0-1 and is assumed to remain constant for the total interval; 2) All decay heat power generated is assumed to be retained in the active core with no heat transfer to the reflector, PCRV internals or primary coolant; and 3) A 10 percent margin has been included on the core heatup time given in Figure 3.0-2. If the active core remains below 760 degrees F, which corresponds to the design maximum core inlet temperature, then there can be no damage to fuel or PCRV internal components, even in the absence of forced circulation of primary coolant helium flow. N

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BASIS FOR SPECIFICATION LCO 3.0/SR4.0 (continued)

The time required to reach a CALCULATED BULK CORE TEMPERATURE of 760 degrees F is primarily dependent upon the decay heat power and the current average core temperature. This time is conservatively estimated using the data in Figures 3.0-1 and 3.0-2. The decay heat power data in Figure 3.0-1 was explicitly calculated for the Fort St. Vrain core and is derived from Appendix D.1 of the FSAR, Figure D.1-9, revision 2. The decay heat power resulting from a varying power history can be conservatively calculated by representing the actual power history by a series of constant power steps and then summing the individual decay heat power contribution from each power step. The decay heat power due to operation during the last 1000 days can be determined in this manner. Residual decay heat power from earlier operation is conservatively estimated by assuming that this was full power continuous operation, and by then adding this decay heat power component to the calculated decay heat power value.

N

Knowing the decay heat power and the current average core temperature, the time for the core to heat up from its current temperature to 760 degrees F can be obtained from Figure 3.0-2, which has been generated using the adiabatic heat transfer model and a heat capacity for composite graphite as given in Appendix D.1 of the FSAR, Figure D.1-3, revision 2.

To allow for uncertainties associated with determining the time to reach a CALCULATED BULK CORE TEMPERATURE of 760 degrees F, an additional 10 percent has been included in the decay heat energy given in Figure 3.0-2. In addition, it has been specified that any time interval for which the primary coolant flow is interrupted shall not exceed 21 days. This ensures a restoration of forced circulation of primary coolant flow to confirm core average temperature on a periodic basis. Although much longer intervals can be determined from Figure 3.0-2, 21 days is an adequate time to conduct operations requiring flow interruption, such as maintenance or circulator changeout. Operating experience at Fort St. Vrain has shown that the calculated core heatup rate has always been higher than the actual core heatup rate.

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REACTIVITY CONTROL SYSTEMS

3/4.1.1 CONTROL ROD PAIR OPERABILITY

LIMITING CONDITION FOR OPERATION

3.1.1 All control rod pairs not fully inserted shall be OPERABLE with:

a. A scram time less than or equal to 152 seconds from the fully withdrawn position, | R

b. A control rod drive (CRD) motor temperature less than or equal to 250 degrees F,

c. A helium purge flow not carrying condensed water to each CRD penetration when reactor pressure is above 100 psia, and

d. The absence of a slack cable alarm.

APPLICABILITY: POWER, LOW POWER, and STARTUP

ACTION:

a. With one or more control rod pairs inoperable due to being immovable (i.e., not capable of being fully inserted) or if the slack cable alarm cannot be cleared, immediately initiate a reactor shutdown and an assessment of the SHUTDOWN MARGIN, and be in at least SHUTDOWN within the next 12 hours.

b. With one control rod pair inoperable due to having a scram time greater than 152 seconds, operation may continue provided that within 24 hours:

1. The control rod pair is restored to OPERABLE status, or

2. The control rod pair is fully inserted, or

3. The SHUTDOWN MARGIN requirement of Specification 3.1.3 is satisfied with the control rod pair considered inoperable in its present position.

If none of the above conditions can be met, be in at least SHUTDOWN within the next 12 hours.

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BASIS FOR SPECIFICATION LCO 3.1.1/SR 4.1.1

Control rod pair OPERABILITY ensures that a minimum SHUTDOWN MARGIN is capable of being maintained.

The control rod pair withdrawal accident analyses described in FSAR Sections 14.2.2.6 and 14.2.2.7 were performed assuming a scram insertion time of 152 seconds and a ramp reactivity insertion of 0.080 delta k and 0.058 delta K, respectively.

Requiring the scram time to be less than or equal to 152 seconds will ensure that the ramp reactivity rate is consistent with that assumed in the accident analyses. The full insertion scram time can be determined either directly from a full insertion scram test or indirectly from a partial scram test of 10 inches or more. For the partial scram test, the estimate of an extrapolated scram time of less than or equal to 152 seconds is always based on assuming a scram from the fully withdrawn position and not from the actual rod position. R

The total calculated reactivity worth of all 37 control rod pairs is 0.210 delta k, which is significantly greater than the scram reactivities assumed in the accident analyses. Therefore, a single control rod pair with a scram time greater than 152 seconds, as allowed in ACTION b of the specifications, will have no impact on the calculated consequences of the control rod pair withdrawal accident.

Temperature Limitation

Control Rod Drive Mechanism (CRDM) qualification tests were performed in a 180 degree F helium environment. The motor and brake were energized and deenergized in severe duty cycles up to once every 5 seconds for 630,000 jog cycles and 5000 scrams of the CRDM. CRDM motor temperatures ranged from 200 degrees F to 230 degrees F with an average of 215 degrees F during these tests. During power ascension testing, CRDM temperatures up to 213 degrees F were experienced at power levels up to 70%. Using data obtained during power ascension testing, a CRDM temperature of 260 degrees F was predicted for 100% power conditions with an orifice valve fully closed. The minimum predicted open position for an orifice valve at 100% power is about 10% for which the predicted CRDM temperature is 250 degrees F. Tests conducted to 100% power indicated these predictions to be conservative because the maximum measured CRDM motor temperature was 218 degrees F. The operating temperature of the CRDM is limited by the motor insulation which is derated for 272 degrees F to account for motor temperature rise, frictional torque increase, and winding life expectancy. See section 3.8.1.1 of the FSAR.

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REACTIVITY CONTROL SYSTEMS

3/4.1.6 RESERVE SHUTDOWN SYSTEM

LIMITING CONDITION FOR OPERATION

3.1.6.1 All reserve shutdown (RSD) units shall be OPERABLE with: |R

a. At least 1500 psig pressure in their individual helium gas bottle supplies, and

b. At least 500 psig pressure in the Alternate Cooling Method (ACM) nitrogen bottles which provide a backup means of actuating the RSD hopper pressurization valves.

APPLICABILITY: POWER, LOW POWER, and STARTUP

ACTION: a. With one RSD unit inoperable, operation may continue provided that an OPERABLE spare RSD unit is available.

b. With two or more RSD units inoperable, restore all but one inoperable RSD unit to OPERABLE within 24 hours, or be in at least SHUTDOWN within the next 12 hours.

c. The provisions of Specification 3.0.4 are not applicable for changes between STARTUP, LOW POWER, and POWER. Prior to entry into STARTUP from SHUTDOWN, all the requirements of this LCO must be met, without reliance on the provisions contained in the ACTION statements.

SURVEILLANCE REQUIREMENTS

4.1.6.1 The reserve shutdown system shall be demonstrated OPERABLE: |R

a. At least once per 7 days by verifying that the pressure of each helium gas bottle is at least 1500 psig.

b. At least once per 7 days by verifying that the pressure of each ACM nitrogen bottle is at least 500 psig.

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c. At least once per 92 days by:

   1. Pressurizing each of the 37 RSD hoppers above reactor pressure, as indicated by operation of the hopper pressure switch,

   2. Operating the ACM quick disconnect couplings, and

   3. Performing a CHANNEL FUNCTIONAL TEST of the instrumentation which alarms at low pressure in the RSD actuating pressure lines.

d. At least once per 366 days by performing a CHANNEL CALIBRATION of the gas pressure instrumentation.

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REACTIVITY CONTROL SYSTEMS

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3/4.1.6 RESERVE SHUTDOWN SYSTEM

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LIMITING CONDITION FOR OPERATION

3.1.6.2 Reserve shutdown (RSD) units on control rod drive assemblies whose control rod pairs are capable of being withdrawn shall be OPERABLE (except RSD units in any control rod drive assemblies removed for refueling/repair) with:

a. At least 1500 psig pressure in their individual He gas bottle supplies.

b. At least 500 psig pressure in the ACM nitrogen bottles which provide a backup means of actuating the RSD hopper pressurization valves.

APPLICABILITY: SHUTDOWN and REFUELING

ACTION: With less than the required RSD units OPERABLE, within 24 hours:

a. Return all control rod pairs (except the ones removed for refueling/repair) to the full-in position, or

b. Verify SHUTDOWN MARGIN requirements are met (LCO 3.1.3), or

c. Insert sufficient RSD material to maintain SHUTDOWN MARGIN requirements.

N

SURVEILLANCE REQUIREMENTS

4.1.6.2 The reserve shutdown system shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that the pressure of each required individual hopper He gas bottle is at least 1500 psig.

b. At least once per 7 days by verifying that the pressure of each required ACM nitrogen bottles is at least 500 psig.

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c. At least once per REFUELING by:

1. Demonstrating that each subsystem is OPERABLE by actuating each group of pressurizing valves from the control room and verifying that the valves open. The capability of pressurizing the corresponding hoppers need not be demonstrated during this test.

2. Performing a CHANNEL CALIBRATION of the RSD hopper pressure switches at the time of control rod drive preventive maintenance (Specification 4.1.1).

3. Visually examining the pipe sections which require disassembly and reassembly within the refueling penetrations, after they have been disassembled for preventive maintenance (Specification 4.1.1), and verifying that there is no deformation or corrosion that could affect RSD system OPERABILITY.

4. Functionally testing two RSD assemblies, removed from the core during the current refueling, out of the core. One assembly shall contain 20 weight % boronated material and the other 40 weight % boronated material. The tests consist of pressurizing the RSD hopper to the point of rupturing the disc and releasing the poison material.

   The absorber material from the tested RSD hoppers shall be visually examined for evidence of boric acid crystal formation and chemically analyzed for boron carbide and leachable boron content. Failure of a RSD assembly to perform acceptably during functional testing or evidence of extensive boric acid crystal formation will be reported to the Commission within 30 days per Specifications 6.9.

d. Following entry of condensed water into any RSD system hopper(s) (see Specification 3.1.1 ACTION h.), by performing the Surveillance Requirements identified in Specifications 4.1.6.2.c.4.

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BASIS FOR SPECIFICATION LCO 3.1.6/SR 4.1.6

The reserve shutdown (RSD) system must be capable of achieving reactor shutdown in the event that the control rod pairs fail to insert.

After extended power operation, the RSD system must add sufficient negative reactivity to overcome the temperature defect between 1500 and 220 degrees F, the decay of Xe-135, the buildup of Sm-149, and some decay of Pa-233 to U-233.

The core reactivity increase due to core cooldown and Xe-135 decay occurs within a few days and was calculated to be between 0.089 delta k and 0.081 delta k, at the beginning and end of the initial cycle, respectively, and about 0.076 delta k for the mid cycle of the equilibrium core. The reactivity increase is largest in the initial core where the thorium loading is high and decreases through the first six cycles to a minimum value for the equilibrium core. The reactivity increase due to Sm-149 buildup and Pa-233 decay occurs over several weeks to months and increases the core excess reactivity for the equilibrium core by about 0.007 delta k during the first 14 days, and by about 0.024 delta k after a few months, including full Pa-233 decay. Therefore, the reactivity control requirement for the RSD system, including an allowance of 0.01 delta k for SHUTDOWN MARGIN, in the absence of any control rod pairs being inserted is 0.098 delta k for the initial core and 0.093 delta k for the equilibrium core after 14 days of Pa-233 decay and 0.121 delta k and 0.110 delta k after full Pa-233 decay. (FSAR Section 3.5.3).

The calculated worth for the RSD system as noted in FSAR Section 3.5.3 is at least 0.13 delta k in the initial core, and 0.12 delta k in the equilibrium core. Although not summarized in the FSAR, the calculated worth for an inoperable RSD unit is about 0.020 delta k, which reduces the total worth to 0.110 delta k, in the initial core and 0.100 delta k in the equilibrium core, which is sufficient to ensure SHUTDOWN during the first 14 days of Pa-233 decay.

Generally, inoperable RSD units are capable of being restored to OPERABLE status within 24 hours. However, in the unlikely event that an inoperable RSD unit cannot be restored to OPERABLE within this time, there is adequate time (at least 14 days due to the slow Pa-233 decay as discussed in the BASIS for Specification 3.1.3) following a shutdown using the RSD system, to allow for corrective action of changing out a CRD assembly. A spare RSD unit is considered available if it is on site

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Ensuring SHUTDOWN MARGIN requirements for a CORE AVERAGE TEMPERATURE greater than or equal to 220 degrees F is acceptable and provides for changing out a Control Rod Drive (CRD) assembly, if necessary. Under normal conditions when the reactor has been operated for several months (which is required for Pa-233 buildup), a CORE AVERAGE TEMPERATURE greater than 220 degrees F is retained for a period of 2-4 weeks even with the CORE AVERAGE INLET TEMPERATURE as low as 100 degrees F. This is adequate time for the replacement of a CRD assembly.

Two or more RSD units may be inoperable for 24 hours to provide a reasonable time for repair. This is permissible because the control rod pairs are available to shut down the reactor in the unlikely event that a shutdown would be required during this short period of time.

A minimum pressure of 1500 psig in the individual helium gas bottle supplies is adequate because the pressure required to burst the rupture discs is 1100 psig (FSAR Section 3.8.3). The rupture discs are designed and have been tested to burst at a differential pressure of 165 plus or minus 50 psi.

A minimum pressure of 500 psig in the ACM nitrogen bottles is adequate because the required set pressure is 220 psig. A set pressure of 220 psig is based on stroking a bank of 10 RSD valves one time and keeping the regulator fully open. This value also compensates for minor line losses and system leakages.

Each of the 37 RSD hoppers shall be pressurized above reactor pressure at least once per 92 days. Two redundant pressurizing valves will be opened using local test switches and the corresponding hopper pressure observed to increase. To prevent releasing absorber material, the high pressure gas cylinder is isolated and the pressurized actuating line is vented prior to the test. Pressurization is accomplished using test gas at a pressure differential of approximately 40-50 psi above reactor pressure, which is below the 115 psi differential pressure required to rupture the disc. The h[illegible] pressure should increase at least 10 psi above reactor pr[illegible]ure, as indicated by the hopper high pressure alarm.

A CHANNEL FUNCTIONAL TEST will be performed on the low pressure alarm instrumentation at least once per 92 days to ensure that the minimum require rupture gas pressure can be monitored.

A CHANNEL CALIBRATION will be performed on the gas pressure instrumentation at least once per 365 days to ensure reliable monitoring of the helium and nitrogen gas supplies.

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In the event that condensed water enters into any RSD system hoppers, (and during each refueling outage) two RSD hoppers shall be functionally tested out of the core. One assembly will contain 20 weight percent and the other 40 weight percent boronated material. The RSD hopper will be pressurized to the point of rupturing the disc and releasing the poison material. The material will be visually examined for boric acid crystallization and chemically analyzed for boron carbide and leachable boron content.

At each refueling, each group of pressurizing valves will be actuated from the control room to verify that the valves open.

At each refueling, the RSD hopper pressure switches which measure the pressure differential between the hoppers and the reactor will be calibrated as individual control and orifice assemblies are removed form the reactor for servicing and maintenance. These switches alarm high pressure for pressurization testing or actual system operation.

The refueling penetration pipe sections will be visually examined for deformation and corrosion following disassembly for refueling or maintenance.

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CORE IRRADIATION, TEMPERATURE AND FLOW LIMITS

3/4.2.1 CORE IRRADIATION

LIMITING CONDITION FOR OPERATION

3.2.1 The maximum in-core irradiation of the fuel elements, control rods, and reflector elements immediately adjacent to the active core shall not exceed the equivalent of 1800 Effective Full Power Days (EFPDs).

APPLICABILITY: POWER, LOW POWER, and STARTUP

ACTION: With the in-core irradiation lifetime of any fuel element, control rod, or reflector element adjacent to the active core exceeding the above limit be in SHUTDOWN within 72 hours.

SURVEILLANCE REQUIREMENTS

4.2.1.1 Prior to entering STARTUP following each refueling, it shall be determined that the in-core irradiation lifetime of all fuel elements, control rods, and reflector elements adjacent to the active core will be less than the above limit for the duration of the next cycle.

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BASIS FOR SPECIFICATION LCO 3.2.1 / SR 4.2.1

The integrity of the fuel particle coatings and graphite dimensional changes is dependent on many variables. Prime variables are the total burnup accumulated by the coated fuel particle and the fast fluence. Limiting the allowable irradiation lifetime to 1800 EFPDs, in conjunction with the peaking factor limits of Specification 5.3, will ensure that the coated fuel particles and graphite will remain within the demonstrated irradiation test values. The burnup and irradiation test results (FSAR Appendix A.2) are generally described in terms of percent Fissions per Initial Metal Atom (FIMA) for both the fissile and fertile particles.

The basis for the design lifetime for the fuel elements, control rods, and replaceable reflector elements is described in Sections 3.2 and 3.8 of the FSAR. For the fuel and reflector elements (FSAR Section 3.2.2.2), consideration is given to mechanical loads and stresses for handling, supporting, earthquakes, thermal and irradiation differentials, and steam reactions during both steady state and transient operation. The elements' integrity will be sufficient to permit safe removal from the core after 1800 EFPDs of operation.

For the control rods (FSAR Sections 3.2.2.6 and 3.8.1.2), this lifetime will ensure that reactivity control is maintained even if the control rods are inserted for the total duration, without any significant loss of absorber worth or structural or functional deterioration.

The core irradiation limit of 1800 EFPDs is related to a residence time in the core for each element. An evaluation of the residence time records and an allowance for the duration of the next fuel cycle will ensure that this Specification is not exceeded during the next cycle of operation for any element. R

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INSTRUMENTATION

3/4.3.2 MONITORING INSTRUMENTATION

SEISMIC INSTRUMENTATION

LIMITING CONDITIONS FOR OPERATION

3.3.2.3 The seismic monitoring instrumentation shown in Table 3.3.2-2 shall be OPERABLE.

APPLICABILITY: At all times

ACTION:

a. With the number of OPERABLE seismic monitoring instruments less than the minimum instruments OPERABLE requirement for more than 30 days, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 10 days outlining the cause of the malfunction and the plans for restoring the instrument(s) to OPERABLE status.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.2.3.1 Each of the seismic monitoring instruments shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION, and CHANNEL FUNCTIONAL TEST operations at the frequencies shown in Table 4.3.2-2.

4.3.2.3.2 Following a seismic event, all of the seismic monitoring instruments shall be restored to OPERABLE status within 24 hours and the calibration of the vertical seismic triggers shall be checked via a CHANNEL FUNCTIONAL TEST within 5 days following the seismic event. For all seismic instruments found out of calibration, a CHANNEL CALIBRATION shall be performed within 30 days following the seismic event. Data shall be retrieved from actuated instruments and analyzed to determine the magnitude of the vibratory ground motion. Upon the actuation of a seismic trigger due to a seismic event, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 14 days describing the magnitude, frequency spectrum, and resultant effect upon unit features important to safety. N

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TABLE 3.3.2-2

SEISMIC MONITORING INSTRUMENTATION

INSTRUMENTS AND SENSOR LOCATIONS	SETPOINT	MEASUREMENT RANGE	MINIMUM INSTRUMENTS OPERABLE
1. Triaxial Time-History Accelerographs and Vertical Seismic Triggers			2
a. PCRV Support Ring*	LTE 0.015g	-1g to +1g	
b. Top of PCRV*	LTE 0.015g	-1g to +1g	
c. Visitors Center	LTE 0.015g	-1g to +1g	
2. Seismoscopes			2
a. PCRV Support Ring	N/A		
b. Top of PCRV	N/A		
c. Visitors Center	N/A		

* With control room alarm

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TABLE 4.3.2-2

SEISMIC MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENTS AND SENSOR LOCATIONS	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	
1. Triaxial Time-History Accelerographs and Vertical Seismic Triggers				
a. PCRV Support Ring**	Q*	18 Mos.	SA	
b. Top of PCRV**	Q*	18 Mos.	SA	
c. Visitors Center	Q*	18 Mos.	SA	
2. Seismoscopes				
a. PCRV Support Ring	M	18 Mos.	N/A	R
b. Top of PCRV	Q	18 Mos.	N/A	
c. Visitors Center	M	18 Mos.	N/A	R

* Except seismic trigger

** With control room alarms

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BASIS FOR SPECIFICATION LCO 3.3.2.3 / SR 4.3.2.3

The OPERABILITY of the seismic instrumentation ensures that sufficient capability is available to determine the magnitude of a seismic event, the response of the facility, and to sound an alarm in the event that a disturbance greater than the setpoint is experienced. (The minimum instruments OPERABLE also reflects the need to send instruments off-site for calibration.) This capability permits comparison of the measured response to that used in the design basis for the facility to determine if plant shutdown or inspection is necessary pursuant to FSAR Section 7.3, and the plant emergency procedures.

The nominal setpoint for the vertical seismic triggers is 0.01g. The 0.015g figure in Table 3.3.2-2 reflects instrumentation calibration tolerances.

The intervals specified for testing and calibration of the seismic instrumentation are consistent with the Standard Technical Specifications for LWR's and industry practice and are, therefore, considered adequate to ensure the instruments operate as intended.

A CHANNEL CHECK will be performed once per 92 days to verify the OPERABILITY of the battery pack and charger. The time-history accelerographs including seismic triggers are sent to the manufacturer for calibration on an 18 month cycle (in accordance with the manufacturer's recommendation). The calibration of the seismic triggers is additionally verified on-site. This on-site calibration verification will also be performed following seismic events which cause the instruments to be actuated. Instruments so determined to be out of calibration will be sent to the manufacturer for CHANNEL CALIBRATION.

The seismoscopes are smoked glass devices wherein vibration causes a needle to etch a trace in the smoked glass. A CHANNEL CHECK consists of shining a flashlight through a viewing port to see if the device has been actuated. For readily accessible seismoscopes, this is performed monthly. The less accessible devices are checked quarterly. N

The Special Report required in Specification 4.3.2.3.2 to be submitted to the Commission following a seismic event does not include response spectra data. Response spectrum data, when deemed necessary, will be obtained by off-site digitization of the film data and subsequent data reduction which requires several weeks. Included in this report will be the resultant effect of the seismic event on the Class I structures, systems, and components as listed in Table 1.4-1 of the FSAR.

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PLANT AND SAFE SHUTDOWN COOLING SUPPORT SYSTEMS

3/4.7.2 HYDRAULIC POWER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.2 Two hydraulic power systems, providing control to their respective coolant loops, shall be OPERABLE with:

a. One OPERABLE hydraulic valve accumulator and associated header servicing each group of valves,

b. Hydraulic fluid pressure to each group of valves maintained greater than 2500 psig,

c. At least two OPERABLE hydraulic pumps, and

d. Hydraulic oil reservoir temperature less than 150 degrees F.

APPLICABILITY: POWER, LOW POWER, and STARTUP

ACTION:

a. With no hydraulic valve accumulator or with loss of capability to supply at least 2500 psig to the valve operators of one group of valves, isolate the affected secondary coolant loop within 1 hour, and be in SHUTDOWN within 24 hours.

b. With the hydraulic oil temperature exceeding 150 degrees F, restore the oil temperature to within its limit within 24 hours or be in SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.7.2 The hydraulic power system shall be demonstrated OPERABLE:

a. At least once per 24 hours by verifying:

1. That hydraulic fluid pressure to each group of valves is greater than 2500 psig, and

2. That hydraulic oil reservoir temperature is less than 150 degrees F.

b. At least once per 18 months by verifying that the standby pump automatically starts when system pressure drops to less than 2800 psig.

BASIS FOR SPECIFICATION LCO 3.7.2 / SR 4.7.2

Each secondary coolant loop hydraulic power system is designed with the following: three hydraulic fluid pumps, two hydraulic accumulators for each group of hydraulic operated valves, and separate headers to each group of valves. The hydraulic system will normally operate with two hydraulic fluid pumps and both hydraulic accumulators in service. The third hydraulic pump and one accumulator in each group is redundant. (FSAR Section 9.11).

The hydraulic oil temperature limit of 150 degrees F corresponds to the system design temperature and minimizes oxidation of the hydraulic fluid, thereby enhancing its service life. The temperature of the hydraulic oil reservoirs, which are immediately downstream of the hydraulic oil coolers, is alarmed in the control room.

Loss of two hydraulic fluid pumps or both hydraulic accumulators servicing a group of valves indicates the potential for complete or partial loss of valve OPERABILITY in the affected secondary coolant loop. A 1 hour ACTION time is provided to isolate the affected loop, in an effort to regain the capability to supply at least 2500 psig and/or the OPERABILITY of at least one accumulator for the affected valve group. If these efforts are not successful, reactor shutdown is required within the next 24 hours. R

In the event hydraulic oil is lost to a group of valves, some degree of control will be lost and the affected secondary coolant loop is isolated. With only one group of valves inoperable, the ability to totally isolate the affected coolant loop is ensured by the selective grouping of valves.

In the event of loss of all hydraulic power in one system, all flow pressure and speed control as well as ability to totally isolate the affected secondary coolant loop is lost. Therefore, the affected loop is isolated with the exception of cold reheat steam path to the condensor via the circulator steam-drive bypass line. Heat removal is accomplished with the non-affected secondary coolant loop. Upon depletion of steam to drive the circulators, the circulator(s) are operated on their Pelton drives.

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Verifying that the standby hydraulic pump starts when system pressure falls below 2800 psig ensures that a system pressure of at least 2800 psig will be maintained at the pumps; this will provide at least 2500 psig at the valve actuators, taking into consideration line losses. This surveillance will be performed once per 18 months. Verifying that hydraulic fluid pressure is greater than 2500 psig once per 24 hours ensures that the minimum pressure required to operate the hydraulic valves in the secondary coolant system is available.

Verifying that hydraulic fluid pressure is greater than 2500 psig once per 24 hours ensures that the minimum pressure required to operate the hydraulic valves in the secondary coolant system is available.

Verifying that the hydraulic oil temperature is less than 150 degrees F once per 24 hours ensures that system oil temperature is within design limits.

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PLANT AND SAFE SHUTDOWN COOLING SUPPORT SYSTEMS

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3/4.7.6 FIRE SUPPRESSION SYSTEMS

HALON SYSTEMS

LIMITING CONDITION FOR OPERATION

3.7.6.3 The following Halon systems and associated HVAC isolation dampers shall be OPERABLE:

a. Control room,

b. Auxiliary electric equipment room,

c. 480 volt switchgear room,

d. Building 10 - switchgear room and ground level,

e. Building 10 - ground level under mezzazine floor, and

f. Building 10 - battery room.

APPLICABILITY: At all times

ACTION:

a. With one or more of the above required Halon systems or HVAC isolation dampers inoperable, within 1 hour establish a continuous fire watch with backup fire suppression equipment for the affected room(s).

b. The provisions of Specification 3.0.3 and 3.0.4 are not applicable.

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SURVEILLANCE REQUIREMENTS

4.7.6.3 Each of the above required Halon systems shall be demonstrated OPERABLE:

a. At least once per 31 days, by verifying pressure in the Halon bottles.

b. At least once per quarter by verifying that the Halon storage tank quantity is at least 95% of full charge and pressure is at least 90% of full charge pressure for the following systems:

1. Building 10 - switchgear room and ground level,

2. Building 10 - ground level under mezzanine floor, and

3. Building 10 - battery room

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c. At least once per 6 months, by verifying that the Halon storage tank weight is at least 95% of full charge weight (or level) and pressure is at least 90% of full charge pressure for the following systems:

1. Control room,

2. Auxiliary electric equipment room, and

3. 480 Volt switchgear room

d. At least once per 18 months, by:

1. Verifying that the system, including associated HVAC isolation dampers, actuates correctly upon receipt of a simulated test signal, and

2. Verifying that the distribution headers and nozzles are not blocked by flowing air through the system.

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BASIS FOR SPECIFICATION LCO 3.7.6.3 / SR 4.7.6.3

The OPERABILITY of the Halon systems ensures that adequate fire suppression capability is available for all postulated fires in the three room control complex and in building 10. The Halon system consists of two main distribution systems, one for building 10 and the other for the three room control complex. The main distribution systems are separated into independent subsections that provide Halon to the following fire areas that house safety related equipment: 1) the control room, 2) the auxiliary electric equipment room, 3) the 480V switchgear room, 4) Building 10's switchgear room and ground level, 5) Building 10's ground level under the mezzazine floor, and 6) Building 10's battery room.

Halon is supplied from full capacity main cylinders and 100% spare reserve cylinders; either the main or the reserve cylinders may be used to satisfy the Specification requirements. The OPERABILITY of the associated HVAC isolation dampers ensures that adequate room isolation will be available to maintain an effective concentration of Halon after actuation of the suppression system. In the event that portions of the Halon suppression systems are inoperable, backup fire fighting equipment is required in the affected areas until the inoperable equipment is restored to service. An installed sprinkler system (Specification 3.7.6.1) provides dedicated backup suppression for the 480 volt switchgear room and the auxiliary electric equipment room.

The surveillance requirements ensure that the minimum OPERABILITY requirements of the Halon suppression systems are met. A semi-annual surveillance ensures that a sufficient volume of Halon is in the storage tanks by verifying either the weight or the level of the tanks. For the building 10 Halon system, storage tank pressure and quantity are verified quarterly, because the storage tank weight cannot be verified without removing the tanks. Quantity is determined by use of a surveillance method which is acceptable to the ANI, such as the heat tape and gun method. The 31 day surveillance for checking pressure in the Halon systems is adequate verification of proper valve lineup as a mispositioned valve results in discharge of the Halon. Verification that the distribution headers are not blocked demonstrates their ability to spray Halon when needed to suppress a fire. Verification that the system, and its associated HVAC isolation dampers react to a simulated actuation signal will ensure overall system response to a postulated fire. R

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3/4.9.1 FUEL HANDLING AND MAINTENANCE IN THE REACTOR

LIMITING CONDITION FOR OPERATION

3.9.1 The following reactor conditions shall be maintained:

a. The PCRV shall be depressurized to atmospheric pressure or slightly below,

b. The CORE AVERAGE INLET TEMPERATURE shall be 165 degrees F or less *, and | D

c. The SHUTDOWN MARGIN requirements of Specification 3.1.3 shall be met.

APPLICABILITY: Whenever both primary and secondary PCRV closures of any PCRV penetration are removed

ACTION:

a. With the conditions of a or b above not met, restore the condition(s) to within the above limits within 1 hour, or terminate fuel handling and vessel internal maintenance, retract the fuel handling mechanism or any other remote operated mechanisms from the PCRV, and close the reactor isolation valve or opening through the PCRV as soon as practicable.

b. With the SHUTDOWN MARGIN requirements of Specification 3.1.3 not met, comply with ACTION c of Specification 3.1.3.

SURVEILLANCE REQUIREMENTS

4.9.1 a. The reactor pressure and temperature conditions shall be determined to be within the above limits at least once per 12 hours, | D

b. Verification of SHUTDOWN MARGIN shall be in accordance with Specification 4.1.3.

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* Applicable only when the fuel handling machine is located on the reactor vessel, with the cask isolation valve and reactor isolation valve open. | D

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BASIS FOR SPECIFICATION LCO 3.9.1 / SR 4.9.1

To prevent the outleakage of primary coolant and potential release of activity during refueling or maintenance in the reactor vessel, the reactor must be depressurized and maintained within the required conditions. The CORE AVERAGE INLET TEMPERATURE is limited to 165 degrees F to prevent short-term pressurization of the fuel handling equipment over 5 psig (the maximum allowable working pressure of the fuel handling equipment) as a result of accidental inleakage of water into the vessel during refueling.

The ACTION statement ensures that reactor and fuel handling machine will be placed in the safest configuration as soon as practicable, if a required condition cannot be maintained.

The Surveillance Requirement frequency gives adequate assurance that changes in reactor conditions will be detected in time to permit corrective actions if required.

FUEL HANDLING AND STORAGE SYSTEMS

3/4.9.2 INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.9.2 The reactivity of the core shall be continuously monitored by at least two startup channel neutron flux monitors.

APPLICABILITY: REFUELING

ACTION:

a. With one of the above required neutron flux monitors inoperable, or not operating, immediately suspend all operations involving CORE ALTERATIONS, any evolution resulting in positive reactivity changes, or movement of IRRADIATED FUEL.

b. With both of the above required neutron flux monitors inoperable or not OPERATING:

1. Immediately suspend all operations involving CORE ALTERATIONS, any evaluation resulting in positive reactivity changes, or movement of IRRADIATED FUEL,

2. Retract the fuel handling mechanism or any other remote operated mechanism from the PCRV,

3. Close the reactor isolation valve or opening through the PCRV as soon as practicable, and

4. Within 12 hours evaluate the SHUTDOWN MARGIN per Specification 4.1.3.

SURVEILLANCE REQUIREMENTS

4.9.2 Each startup channel neutron flux monitor shall be demonstrated OPERABLE by performance of:

a. A CHANNEL CHECK at least once per 24 hours,

b. A CHANNEL FUNCTIONAL TEST within 24 hours prior to the initial start of CORE ALTERATIONS, and

c. A CHANNEL FUNCTIONAL TEST at least once per 7 days.

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BASIS FOR SPECIFICATION LCO 3.9.2 / SR 4.9.2

The OPERABILITY of the neutron flux monitors ensures that redundant monitoring capability is available to detect changes in the reactivity condition of the core. (Additional information is contained in the BASIS for Specification 3.3.1).

The ACTION statement ensures that activities that could affect the reactivity condition of the core are suspended whenever neutron flux monitoring capabilities are degraded.

The Surveillance Requirements assure that the neutron flux monitors are capable of detecting changes in reactor conditions in time to permit corrective actions if required. These are in addition to the calibration requirements of SR 4.3.1.

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Alternates

6.5.1.3 An alternate chairman and alternate members, if required, shall be appointed in writing by the PORC Chairman to serve in the absence of a chairman or a member; however, no more than two alternate members shall participate in PORC activities at any one time.

Meeting Frequency

6.5.1.4 The PORC shall meet at least once per calendar month and as convened by the Chairman or his designated alternate.

Quorum

6.5.1.5 A quorum shall consist of the Chairman or alternate Chairman, and four members including alternates.

Responsibilities

6.5.1.6 The PORC shall be responsible for:

a. Review of all procedures required by Technical Specification 6.8.1, 6.8.2, 6.8.3, and 6.8.4 and changes thereto, and any other proposed procedure or changes to approved procedures as determined by the Station Manager to affect nuclear safety. R

b. Review of all proposed tests and experiments that affect nuclear safety.

c. Review of all proposed changes to the Technical Specifications.

d. Review of all proposed changes or modifications to plant systems or equipment that affect nuclear safety.

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least every 3 months until all three events have been completed.

Annual Reports and Summary

6.9.1.2 Annual reports covering the activities of the unit for the previous calendar year shall be submitted as described below. Reports required on an annual basis shall include:

a. Annual Occupations Exposure Report

1. A tabulation on an annual basis of the number of station, utility, and other personnel (including contractors) receiving exposures greater than 100 mrem/year and their associated man-rem exposure according to work and job functions*, e.g., reactor operations and surveillance, in-service inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling shall be submitted by March 31 of each year. The dose assignment to various duty functions may be estimates based on pocket dosimeter, TLD, or film badge measurements. Small exposures totaling less than 20% of the individual total-dose need not be accounted for. In the aggregate, at least 80% of the whole body dose received from external sources shall be assigned to specific major work functions. |e

* This tabulation supplements the requirements of 10 CFR Part 20.407.

c. The results of specific activity analyses in which the primary coolant exceeded the limits of Specification 3.4.1.1 shall be submitted by March 31 of each year. The following information shall be included: (1) Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded (in graphite and tabular format), (2) Results of the last isotopic analysis for radioiodine performed prior to exceeding the limit, results of analysis while limit was exceeded and results of one analysis after the radioiodine activity was reduced to less than limit. Each result should include date and time of sampling and the radioiodine concentrations; (3) Clean-up flow history starting 48 hours prior to the first sample in which the limit was exceeded; (4) Graph of the I-131 concentration (μCi/gm) and one other radioiodine isotope concentration (μCi/gm) as a function of time for the duration of the specific activity above the steady-state level; and (5) The time duration when the specific activity of the primary coolant exceeded the radioiodine limit. R

b. Annual Radiological Environmental Monitoring Report

A report on the Radiological Environmental Monitoring Program for the previous calendar year shall be submitted to the Regional Administrator of the Nuclear Regulatory Commission Regional Office (with a copy to the Director, Office of Nuclear Reactor Regulation) by May 1 of each year.

ATTACHMENT 4

TO P-88045

PCRV SUPPORT RING

TEMPERATURE LIMIT

NRC Comment:

PSC Action 2 in Enclosure 3 to NRC letter from Hunter to Lee dated 8/22/85 (G-85354) states:

> "PSC will evaluate the need for an LCO on PCRV support ring temperatures separate from the upgrade program."

PSC Response:

PSC does not believe that a Technical Specification limit on maximum PCRV support ring temperature is required. Per the guidance provided in ANS 58.4, Section 4.2.4.1, Technical Specifications should be provided for parameter limits "when they are relied upon in the safety analysis."

PSC considers that the PCRV support ring itself is relied upon but its concrete surface temperature does not have to be specified as a condition of operability. A maximum PCRV support ring average concrete temperature of 120 degrees F is identified in FSAR Section E.25. PSC considers that this is provided as part of the design description and not as an operating limit that is relied upon to assure proper support ring function.

FSAR Section 6.2.3.2.5 discusses the fact that the ambient air temperatures within the PCRV support ring have regularly exceeded the values in the design criteria (FSAR Section E.25), but no evidence of adverse effects has been detected after several years of operation under these conditions. PSC continually monitors and controls the temperature within the PCRV support ring, outside of the Technical Specifications. Any significant temperature excursions would be detected and the impact would be assessed.

PSC considers that the PCRV support ring concrete is not significantly different from the PCRV concrete, where temperatures of 150 degrees F (and localized higher temperatures of 250 degrees F) have been evaluated in FSAR Section 5.4.5.3 as not significantly affecting the concrete properties. Also, concrete strength increases over time, so any degradation would most likely still result in concrete strengths that exceed the initial requirements.

The area temperatures that have historically exceeded the design criteria have not approached the temperatures at which concrete degradation would be expected to occur, and PSC considers that monitoring via administrative controls, outside the Technical Specifications, is appropriate.

Proposed Resolution: B