Date: February 10, 1984

To: John Larkins, NRC

From: W. H. McCulloch, 6445

On February 6-7, 1984 Bill McCulloch, John Aragon, and Don King of SLA visited the NTS Hydrogen Burn Facility to inspect the condition of equipment and cable/splice samples after a series of hydrogen burn tests conducted by EPRI. Also present for the inspection were Jack Haugh and George Sliter of EPRI. Dick Miller of Westinghouse, Don Randall of Astron, and John Wanless of the NUS Corporation

The calorimeters we had supplied were of special interest to us, but, at the request of John Larkins, NRC, we also inspected the other equipment and cable/splice samples from their original (new) condition, and we made no attempt to evaluate or interpret the observed conditions of the samples or to assess their operability or reliability. The equipment and samples were tagged and photographed and the condition of each item was noted in an Engineering Record book. The inspection consisted of visually examining the exterior of the equipment and cable/splice samples and the interior of the equipment as their covers were removed. At times unusual or unexpected "flakes", "crusts" or corrosion were removed from the equipment and placed in sample bags.

The equipment inspected included absolute and differential pressure gages, solenoid valves, limit switches, resistive temperature devices, penetration assemblies, a fan motor, and a motor operated valve. Cable samples were obtained from various manufactures: Kerite, Rockbestos, Samuel Moore, Raychem, Okonite, Boston Insulated Wire & Cable and Anaconda. Most of the cable samples had been spliced into loops using a variety of splicing techniques not all of the type used in nuclear facilities. We are to be informed later by EPRI which were the higher grade splices.

In general, the equipment exteriors were scratched, discolored and corroded, evidence of their handling and exposure to fires and high temperatures. There were not however, any indications of external damage to the equipment. Upon removing the covers we found water condensed on most surfaces inside the pressure gages. The most probable source for this water intrusion is through the feed throughs for the experimental instrumentation (which is not typical of installation in nuclear facilities). But there was no way to assure that the "nuclear grade" gaskets and seals did not allow the intrusion of at least some of the water. (It should be noted that "nuclear grade" installation procedures and checks must be meticulously followed: other techniques judged acceptable even by experienced and careful installers, can lead to problems as occured repeatedly in this test series.) Except for the condensation and small amounts for foreign material (scale from evaporated water and bits of construction material, e.g., metal filingss) we found no evidence of damage or malfunction. The presence of the water might give some concern if the component were expected to perform over an extended period. There was enough water present to cause shorting in a printed circuit board if the potting material were absent or compromised.

Damage to the cable/splice samples ranged from virtually none to severe damage to the cable jacket. Except for some splices which may not have been "nuclear grade" our visual inspection revealed no evidence of failure of the conductors or insulators. However, there was in many cases sufficient damage to the cable outer jacket to seriously doubt its ability to protect the inner cable parts from a wet environment. For instances, if the inner insulation shrank back slightly from the splices, moisture could easily cause shorts. In the absence of defined criteria for survivability, we cannot say that we observed success or failure. Of approximately fifty cable samples, only two were virtually free of damage. Several showed slight surface melting and a few were charred without significant surface wrinkling or bulging. On many samples the jacket had expanded (either from its own phase change or from pressure generated by high temperture gases inside the jacket) and collapsed giving a heavily wrinkled or knurled appearance (like the rough bark on a mature tree). In the more severe cases the jackets had holes and/or splits to reveal the inner layers. All but one of the splices showed some damage (usually substantial shrinkage or splitting of the splice jacket). Again, our observations were limited primarily to the cable jacket and we saw no clear indication of damage to internal parts or of a cable's inability to perform its function.

THE PRODUCTION OF H_2 IN LWR SYSTEMS DURING SEVERE ACCIDENTS

The early production of H_2 in LWR systems during a severe accident is mainly caused 2by the oxidation of the Zircaloy cladding by steam. This reaction produces one H_2 molecule for every steam molecule reacted. The geometry of an undamaged (i.e., unmelted) core is very susceptible to oxidation by steam at temperatures above 2200^{0}F (the current Appendix K limitation for LOCA events). The very high surface to volume ratio of the Zircaloy cladding allows sufficient contact area for the oxidation to occur.

When an LWR undergoes a severe accident leading to loss of coolant, it eventually reaches the stage at which the primary water is gradually boiled away leaving the core uncovered. For rapid blowdown events such as a large-break LOCA, accumulator injection will refill the core. If the low pressure injection system then fails, the core water will boil off slowly at ambient pressure. Boilaway times from the top of the core to the bottom are of the order of 20-40 minutes depending on the exact sequence <u>if no additional water is injected into the core by operator action</u> (Note that at TMI-2 water <u>was</u> injected periodically by operator action.)

State of knowledge calculations and direct experimental data (PBF Severe Fuel Damage Series I, Ex-Pile experiments at KfK in the FRG*) show that for an unattenuated boilaway only 15-25% of the Zircaloy cladding will be fully oxidized prior to gross melting of the core (i.e., when temperatures in excess of 5000^{0}F are reached, and failure of the lower core support plate and lower plenum occur because of the high heating rate incurred.) After gross melting and core collapse into the reactor building lower cavity, the remaining unoxidized Zircaloy may undergo further oxidation depending upon

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ENCLOSURE "1"

the availability of oxidant (steam) and the ability of the Zircaloy (present now with a considerably lower surface to volume ratio within a large mass of molten UO_2, ZrO_2, steel, etc.) to come in contact with the steam. Very little information is available on the oxidation rate in such a system but experiments are planned to obtain the needed data.

For accident sequences where an unattenuated boiloff is prevented by periodic water injections (bleed and feed), somwhat more oxidation of the Zircaloy can be achieved prior to or without gross melting of the core. A situation similar to this occurred at TMI-2 where ~45-50% of the cladding was oxidized. However, there is a limit to the total oxidation possible for this type of sequence also. Maximum oxidation (i.e., maximum H_2 production) will occur if the core is boiled down to certain water levels and held there by feed and bleed operations. However, the water level has to be low enough for considerable oxidation but high enough to provide a sufficient steaming rate to preclude gross melting of the core. Calculations included by the Rogovin Study Group in their report on Three-Mile Island (Volume 2, Part 2, Appendix II.8) illustrate the previous point. Figures 1A, 1B, & 1C show core temperatures achieved for uncoveries of 7-feet, 8-feet, and 9-feet respectively for the TMI-2 core. The individual curves on each plot indicate the axial location of the position from the <u>top</u> of the core. That is, 0 = top of core, 1 = one foot down from top, 2 = two feet down from top, etc. All calculations assumed a boildown to the indicated level in

20 minutes and holding at that level (presumably by bleed & feed) for an additional 60 minutes. To interpret these curves in terms of the production one must understand that very rapid oxidation of Zircaloy will <u>not</u> occur until the Zircaloy temperature exceeds 2200°F. Using the kinetic equation for oxidation of Zircaloy given in NUREG/CR-0497, TREE-1280, Revision 2, one can calculate the percentage of cladding oxidized in one hour at various temperatures. Table 1 shows the results at three temperatures: 1500°F, 1830°F and 2200°F. Note that below 1500° only 2% oxidation will occur in one hour. Since the kinetics are a function of the square root of the time, only 2.8% oxidation will occur after 2 hours, etc. Therefore, for those regions of the core below say 1500°F, it can be safely assumed that very little H_2 will be produced (~less than 3%).

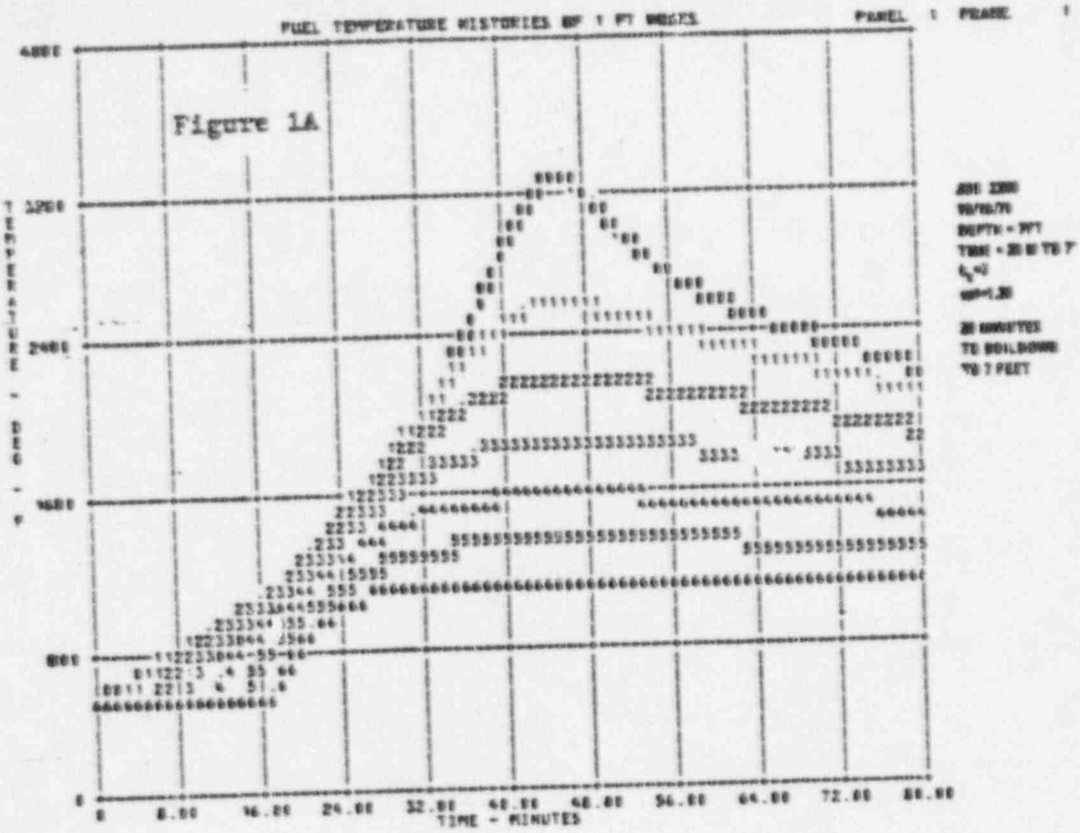
Figure 1A shows, therefore, that for seven feet of the core uncovered for one-hour ~4 feet of the cladding is potentially totally oxidizable. That is, a total H_2 production of only 4/12 x 100 = 33%. For eight feet of the core uncovered (Figure 1B), 5 feet of the core could be totally oxidized leading to a total possible H_2 production from the clad of 5/12 x 100 = 42%. This latter case is considered to be close to what actually happened at TMI-2. Note that for the above two situations no part of the core exceeded 4400°F which is above the cladding melting of point but below the fuel (UO_2) melting point of just over 5000°F. Thus, no gross melting of the core would have taken place except for some dissolution of UO_2 into the liquid Zircaloy (in the upper two feet only).

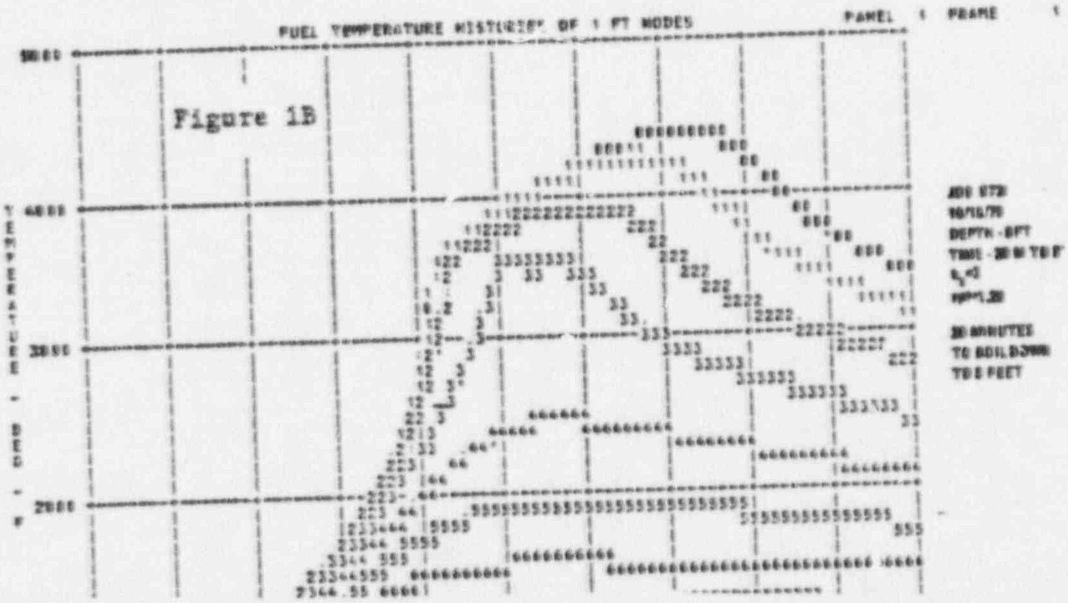
Figure 1C shows the results for a nine-foot uncovery and hold. In this case 7/12 x 100 = 60% H_2 would be produced. However, note that 3.5 feet of the core will now reach the UO_2 melting point and core collapse and gross melting will begin. This large mass may cause failure of the lower plate and lower plenum and lead to a "core on the floor" event. That is, the core would not be recoverable. Note also that even this very severe event oxidized only ~60% of the cladding prior to core collapse.

Therefore, it seems highly unlikely that an event can be constructed to result in oxidation of more than about 50-60% of the cladding and still maintain the core in a recoverable state (i.e., no gross melting and relocation). Experimental data to confirm such analyses will be performed on full-length fuel bundles by NRC in the NRU reactor in Canada over the next year. Follow-on tests will be needed to confirm the calculations for various uncovery levels.

Table 1. Percent Cladding Oxidized at Various Temperatures in One Hour

Temperature (^oF)		Present Clad Oxidized
2200	(1200^oC)	20.5
1830	(1000^oC)	7.8
1500	(800^oC)	2.1


FUEL TEMPERATURE HISTORIES OF 1 FT NODES
PANEL 1 FRAME 1
Figure 1A
TEMPERATURE - DEG - F
TIME - MINUTES
0
8.00
16.00
24.00
32.00
40.00
48.00
56.00
64.00
72.00
80.00
20 MINUTES
TO BOILDOWN
TO 7 FEET


FUEL TEMPERATURE HISTORIES OF 1 FT NODES
PANEL 1 FRAME 1
Figure 1B
TEMPERATURE - DEG - F
TO 8 FEET

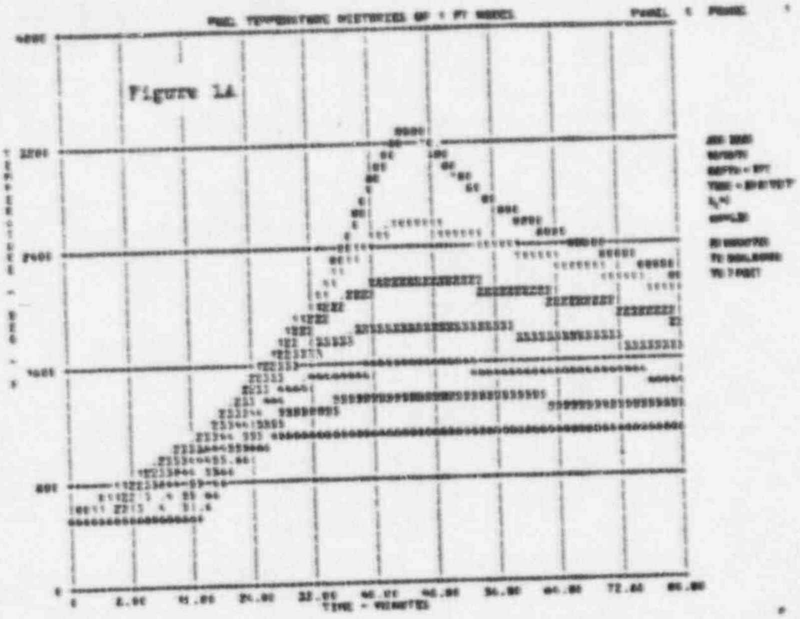


Figure 1A
TIME - MINUTES

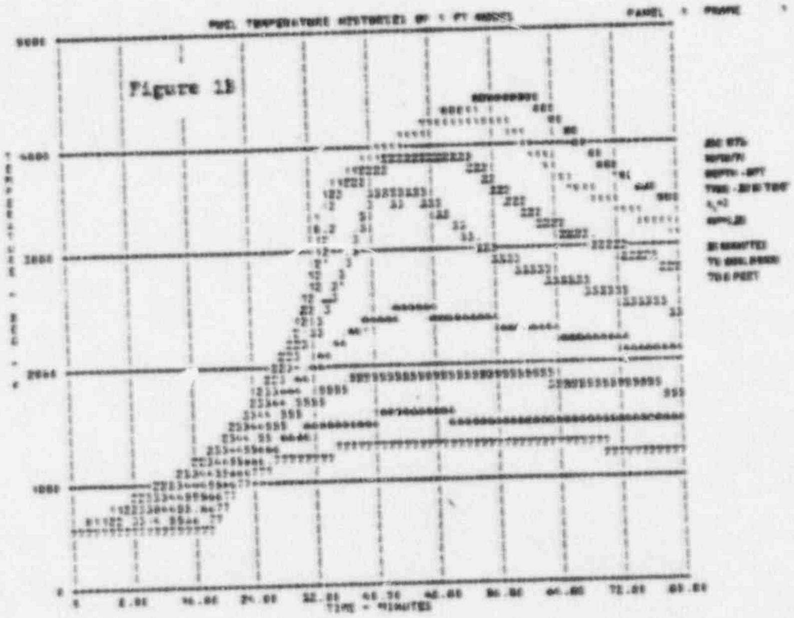


Figure 1B
TIME - MINUTES

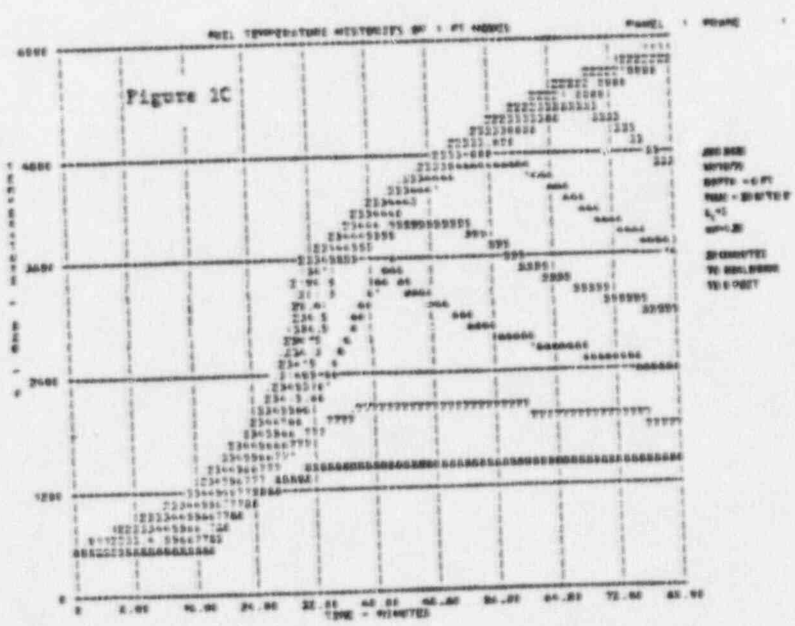


Figure 1C
TIME - MINUTES

Accident Likelihood and Potential for "Arrested" Sequences in PWRs with Large, Dry Containments

Since the time of the original staff proposals on interim hydrogen rulemaking (SECY 81-245A), a considerable amount of analysis has been performed with respect to the likelihood of severe accident sequences in PWRs with large dry containments. A number of PRAs have been performed under the sponsorship of both the NRC and utilities, and have identified those sequences which contribute most importantly to the respective estimates of the frequency of core melting. Such information has been catalogued and studied as one task of the RES-sponsored Accident Sequence Evaluation Program (ASEP). Based on this catalogue, it has been determined that transient-initiated accidents and small LOCAs are, in general, the most important types of sequences in most PWRs. More specifically, ASEP identified five accident sequences which are generally important to the characterization of the core melt frequency in PWRs. These sequences are:

- Transient without early emergency core cooling
- Transient-induced LOCA without early emergency core cooling
- Small pipe break LOCA without early emergency core cooling
- Transient without reactor shutdown (ATWS)
- Transient without early emergency core cooling and without containment heat removal.

ENCLOSURE "J"

An important characteristic of these sequences (except ATWS) is that the loss of water from the reactor coolant system is rather gradual, and thus the processes of core uncovery, core damage, and gross core melting can be protracted. For this reason, there can be a significant potential for operator intervention to restore cooling of the fuel. In general, this recovery potential has not been included in published PRAs. One exception to this is the incorporation of the recovery of offsite electric power during a station blackout accident sequence. In this sequence, a recovery probability (as a function of time) based on historical data is typically included.

The potential benefit of accounting for operator recovery actions has been evaluated as part of the NRC-sponsored IREP PRA of the Arkansas Nuclear One (ANO) Unit 1 plant. In this study, faults which by themselves or in combination with other faults could lead to failures of important systems were categorized as recoverable or non-recoverable (e.g., an inadvertently closed valve was recoverable, a plugged valve was not). For each recoverable fault, a recovery location was identified and a critical time developed, based on how quickly the specific associated function had to be restored. The latter time was based on the onset of an "unacceptable" core condition. For the IREP study, the unacceptable condition was specified as core uncovery, which for transient events and small LOCAs occurred in roughly forty minutes. Using this data and general data on the probabilities of human restoration actions as functions of time, a recovery probability was developed for each recoverable fault.

Since the publication of the ANO IREP, this analysis has been extended to consider restoration times correlated to an unacceptable core condition defined as onset of core melting, rather than core uncovery. In effect, this extended the recovery time for small LOCAs and transients (including loss and subsequent recovery of offsite power) from roughly forty minutes to a time on the order of seventy minutes. Table 1 displays the results of these analyses.

In addition to these assessments of recovery potential, the ANO IREP also identified two accident sequences in which it was possible that the operable core cooling equipment might be adequate to prevent gross core melting, even though it would apparently not prevent core uncovery. In Table 1, the last column ("alternative success criterion") includes this possible effect for the two specific sequences.

The table 1 results suggest several conclusions. With respect to operator recovery actions, it appears that, for this ANO case study:

- operator recovery actions <u>before</u> core uncovery can have a significant effect on estimates of core melt frequency; and

- if recovery actions have not occurred by the time of core uncovery, it is unlikely that such actions will be taken before the onset of gross core melting.

The second conclusion results from two factors. First, many recoverable faults can be rectified very quickly, and, as such, are accounted for in the pre-core uncovery phase (thus the first conclusion). As time proceeds, however, the incremental changes in probability due to recovery actions decrease rapidly. That is, the recovery model corresponds to the intuitive concept that if the operators have been unable to correctly diagnose and account for initial faults by the time the core is first uncovered, then the likelihood of such actions by the time of gross core melting is relatively low. In addition, as such credit for recovery actions decreases the effect of <u>recoverable</u> faults, <u>non-recoverable</u> faults become increasingly more important and finally prevent any further gain. Thus, as Table 1 shows, the only incremental reduction in probability occurs in the station blackout sequence $(T(LOP)\ \overline{AFW}\ \overline{ECC}\ \overline{SPRAYS}\ \overline{RBC})$, due to the somewhat greater likelihood of offsite power restoration prior to core melting. The inclusion of the probability reduction due to an alternative success criterion for two sequences has a more pronounced effect, but the overall reduction still remains rather small.

In summary, for this case study, it can be concluded that recovery actions are a significant factor in preventing severe core damage. However, such actions are most likely to occur before any damage has occurred (i.e., before core uncovery). If the accident has progressed to the point of initial core uncovery, then, under the assumptions used for this case study, it has relatively little chance of being arrested before gross core melting.

TABLE 1

EFFECT OF RECOVERY ASSUMPTIONS ON DOMINANT ACCIDENT SEQUENCE FREQUENCIES

	FREQUENCY (PER R. YR.)			
SEQUENCE	NO RECOVERY	RECOVERY < CORE UNCOVERED	RECOVERY < CORE MELT	ALTERNATIVE SUCCESS CRITERION
SLOCA (1.2) $\overline{\text{ECC}}$	2E-5	3E-6	3E-6	
SLOCA (1.2) $\overline{\text{ECC}}$ $\overline{\text{SPRAYS}}$	2E-5	4E-6	4E-6	
SLOCA (1.7) $\overline{\text{ECCR}}$	3E-5	1E-6	1E-6	
SLOCA (4) $\overline{\text{ECCR}}$	4E-5	1E-6	1E-6	
T(A3) $\overline{\text{S/RV}}$ $\overline{\text{AFW}}$ $\overline{\text{ECC}}$	7E-6	3E-6	3E-6	1 E-7
T(DO1) $\overline{\text{S/RV}}$ $\overline{\text{AFW}}$ $\overline{\text{ECC}}$	2E-5	4E-6	4E-6	1 E-7
T(LOP) $\overline{\text{AFW}}$ $\overline{\text{ECC}}$ $\overline{\text{SPRAYS}}$ $\overline{\text{RBC}}$	4E-5	1E-5	6E-6	
T(FIA) $\overline{\text{RPS}}$ $\overline{\text{ECC}}$	3E-6	3E-6	3E-6	
T(A3) $\overline{\text{AFW}}$ $\overline{\text{ECC}}$	6E-6	1E-6	1E-6	
T(A3) $\overline{\text{AFW}}$ $\overline{\text{ECC}}$ $\overline{\text{SPRAYS}}$	3E-6	1E-6	1E-6	
T(DO1) $\overline{\text{AFW}}$ $\overline{\text{ECC}}$	2E-5	2E-6	2E-6	
T(DO1) $\overline{\text{AFW}}$ $\overline{\text{ECC}}$ $\overline{\text{SPRAYS}}$	1E-5	2E-6	2E-6	
T(DO1) $\overline{\text{AFW}}$ $\overline{\text{ECC}}$ $\overline{\text{SPRAYS}}$ $\overline{\text{RBC}}$	5E-5	3E-6	3E-6	
T(DO2) $\overline{\text{AFW}}$ $\overline{\text{ECC}}$ $\overline{\text{SPRAYS}}$ $\overline{\text{RBC}}$	6E-6	3E-6	3E-6	
ALL DOMINANT SEQUENCES	3E-4	4.1E-5	3.7E-5	3 E-5

TABLE 1 (continued)

TERMS

SLOCA (1.2)	Small LOCA (.4" - 1.2")
SLOCA (1.7)	Small LOCA (1.2" - 1.7")
SLOCA (4)	Small LOCA (1.7" - 4")
T(LOP)	Transient initiated by loss of offsite power
T(A3)	Transient initiated by failure of 4160 VAC bus A3
T(D01)	Transient initiated by failure of 125 VDC bus D01
T(D02)	Transient initiated by failure of 125 VDC bus D02
T(FIA)	Transient with all front-line systems (ECC, AFW, etc.) initially available
$\overline{\text{ECC}}$	Failure of emergency core cooling system in injection mode
$\overline{\text{ECCR}}$	Failure of emergency core cooling system in recirculation mode
$\overline{\text{SPRAYS}}$	Failure of containment spray system
$\overline{\text{S/RV}}$	Stuck-open safety/relief valve
$\overline{\text{AFW}}$	Failure of auxiliary feedwater system
$\overline{\text{RBC}}$	Failure of containment fan cooling system

ENCLOSURE 4 357B

COMMENTS TO
DICK CLEVELAND
7/13/84

For: The Commissioners

From: William J. Dircks
Executive Director for Operations

Subject: STATUS OF HYDROGEN CONTROL ISSUE AND RULEMAKING RECOMMENDATIONS IN SECY-83-357-A

Purpose: To inform the Commission of the background and basis for the proposal that rulemaking with regard to hydrogen control for LWRs with large, dry containments can be safely deferred.

Discussion: In my memorandum of March 15, 1984, (see Enclosure "A"), I informed you of recent test information developed at the Nevada Test Station (NTS) and indicated that it warranted further consideration prior to your decision on the Hydrogen Control Rule (SECY-83-357). The two major issues concern actions to be taken on the Mark III BWRs and ice condenser PWRs, included in the Hydrogen Control Rule, and on the LWRs with large, dry containments. With this Information Paper, the staff provides the background and basis for the proposal that rulemaking with regard to hydrogen control for LWRs with large, dry containments can be safely deferred pending completion of the ongoing NRC- and industry-sponsored research programs.

Since my March 15, 1984 memorandum was written, the staff, in conjunction with industrial participants, has completed a preliminary review of the NTS test data. A summary and description of preliminary results from the NTS program is given in Enclosure "B". The program involved a series of hydrogen combustion tests at premixed hydrogen concentrations varying from 5 to 13 volume percent hydrogen, with different amounts of steam present, as well as tests in which hydrogen was continuously injected at different rates. As indicated in my March 15, 1984 memorandum, the electrical cables appeared to suffer damage during the tests at the higher hydrogen concentrations. Post test examination tests have since been performed on the electric cables to determine the extent of this damage. The staff review of the results of the post test examination of the cables is given in Enclosure "C". Because the cables were not energized during the hydrogen burns, it is not possible after the fact to

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443-7616

with the exception of two cables as noted in Enclosure e

demonstrate conclusively that the cables would have functioned during the burn. However, while there was clearly external damage to cables that burned at both 10 and 13 volume percent hydrogen, the post test examination revealed that the class 1E cables were still able to function. As indicated in Enclosure "C", the cable test procedures included immersion of the cables in water while AC and DC voltage was applied. Some of the cables in the NTS tests had jackets that were damaged in a similar manner to those at TMI-2 which also passed subsequent electrical tests.

In addition to the post test examinations discussed above, the staff has performed its own preliminary review and analysis of the NTS test data and has had many discussions with EPRI regarding the interpretation of the data. The staff has concluded that the NTS experiments were not a ~~direct demonstration of~~ equipment survivability in the ~~ice~~ condenser PWRs and Mark III BWRs because of differences between the test conditions and actual plant conditions. As an example, The hydrogen control system being required by the Hydrogen Control Rule and ~~by current safe~~ practice will effectively limit [redacted] combustion of premixed volumes to hydrogen concentrations of roughly 5 to 9 percent. There was no significant cable burning below 10 volume percent hydrogen in the NTS tests. Equipment failures at lower hydrogen concentrations in the NTS tests are believed by EPRI to be the result of "system" failures, such as improper installation, and were not related to hydrogen combustion. The NTS data should, therefore, prove to be most useful not for a direct indication of equipment survivability, but as a validation of the analytical models that are being used to predict hydrogen burn phenomena and equipment response. No information has been uncovered to date which would lead the staff to conclude that the requirements of the Hydrogen Control Rule could not be reasonably met by the ice condenser PWRs and Mark III BWRs. The NTS data is still undergoing intensive review and additional testing and analysis is planned. The results are being closely monitored regarding their implications relative to the Hydrogen Control Rule.

To put the NTS tests into perspective with regard to the Hydrogen Control Rule now before you, the hydrogen concentration resulting from the oxidation of 75% of the cladding in a PWR with a large, dry or subatmospheric containment would vary from approximately 9.7% for a plant like Zion to 14.8% for a plant like Surrey-3. These figures for volume percent hydrogen do not include the effects of any steam which would be released during the accident raising the containment pressure and reducing the hydrogen concentration. Hydrogen is assumed to be homogeneously mixed when calculating the volume percent, however, the issue of hydrogen stratification is receiving study in the research program. Recent analyses of the accident at TMI-2 estimate the total hydrogen released at approximately 900-1,000 lbs or the equivalent of 45-50% clad-oxidation. This number probably approaches an upper limit for a degraded-core accident, that is, higher clad oxidation percentages are only to be found with core-melt accidents. Enclosure "D" provides the current staff understanding relative to hydrogen production in LWR systems during severe accidents. It is seen that it would be unlikely to have more than about 60% clad-oxidation prior to gross melting of the core. Furthermore, for this to occur, it requires a feed and bleed type of situation to maximize oxidation while preventing full core melting. The water level is thus maintained at some optimum position for maximum hydrogen production, as occurred during the TMI-2 accident. If no water is injected and the core proceeds directly to meltdown, it is expected that significantly less than 60% clad-oxidation would take place (see Enclosure "D"). Recently completed PRA studies on PWRs with large, dry containments (Enclosure "E") indicate that, notwithstanding what occurred at TMI-2, it is relatively unlikely that recovery will take place once initial core uncovery has commenced. Thus, while in the hydrogen rule the staff has conservatively assumed a 75% metal-water reaction for the purposes of design of the hydrogen control system and equipment qualification, the staff believes that it is unlikely that a 75% clad-oxidation can occur. Therefore, we still believe that a 75% clad-oxidation provides a conservative basis for the rule.

Our proposals with regard to the two issues or classes of reactors are as follows:

1. Regarding the Mark III BWRs and ice condenser PWRs, SECY-83-357A recommends that the Commission approve the Hydrogen Control Rule.

2. On page 4 of Enclosure "F", SECY-83-357A, the third sentence in the middle paragraph has been modified so as to read (new words are underlined), "Dry containment designs have a greater inherent capability to accommodate large quantities of hydrogen because of their high design pressure and large volume; therefore, for these designs the Commission believes that rulemaking with regard to hydrogen control can be safely deferred pending completion of NRC- and industry-sponsored research which includes studying the effects of hydrogen burning ~~at higher concentrations~~ to ~~determine~~ effects on equipment survivability."

Simulated

We will continue to monitor the evaluation of the NTS data concerning any implications regarding equipment survivability.

Regarding LWRs with large, dry containments, the primary concern, based on the NTS data, is with hydrogen combustion at high concentrations. Because of the preliminary review indicating that ~~most~~ much equipment could survive a burn at high concentration and because of the small likelihood that a high concentration of hydrogen could be achieved, we recommend that the Commission defer action on the large, dry containment LWRs until completion of the NRC ~~and industry~~ experimental and analytical programs currently underway. The NRC program is intended to supplement the NTS experiments and provide further experimental data for determing equipment survivability in event of a hydrogen burn. Heat transfer to equipment and the resulting temperatures measured at locations in the dewar of the NTS experiments are to be evaluated this summer as EPRI completes their data reduction and makes ~~normalized plots~~ such available to the NRC staff. The results of ~~[illegible]~~ post test analysis will be used to validate or revise the thermal models in the HECTR and HYBER Equipment Response Codes used for predicting equipment temperatures in a hydrogen burn. Separate effects experiments are to be run by the NRC at the Sandia solar radiation facilities exposing cables and equipment to time dependent heat fluxes, environments, and initial temperatures simulating those measured in the NTS experiments. The solar facility will then be used to expose energized cables and active equipment in prototypic configurations to a heat flux predicted for a hydrogen burn in a full-scale containment compartment. Additional tests will be made with cables and equipment to assess the protective factor provided by enclosing cables in conduits, trays or using heat shields; the cooling provided by containment sprays; and the safety margin between the heat flux at which equipment fails or malfunctions and that predicted for a hydrogen burn in a full-scale containment. The code validate or development and separate effects experimental results will provide input to ~~[illegible]~~ Confirm analytical models used to evaluate the thermal response of equipment. The separate effects tests are scheduled to be completed in February 1985 and the analytical evaluations for equipment survivability in full-scale containments, using validated codes, by the end of May 1985.

I will continue to keep you advised of any significant developments related to the hydrogen control program and will provide recommendations for rulemaking related to LWRs with large, dry containments at the conclusion of the ongoing research in early 1985.

William J. Dircks
Executive Director for Operations

Enclosures:
"A" - Memo, dtd 3/15/84, W. J. Dircks to
Commissioners (w/o attachments)
"B" - Summary of Preliminary Results from the
EPRI Nevada Test Station Program
"C" - Memo, dtd 5/16/84, W. S. Farmer to Distribution,
Report of May 10 Meeting with EPRI and Electro-Test
to Review Hydrogen Burn Cable Post-Test Examination
Results
"D" - The Production of H_2 in LWR Systems During Severe
Accidents
"E" - Accident Likelihood and Potential for "Arrested"
Sequences in PWRs with Large, Dry Containments

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WButler, NRR
ZRosztoczy, NRR
RCleveland, NRR
CTinkler, NRR
HGarg, NRR
VNoonan, NRR
RCarter, NRR
RLaGrange, NRR
VBenaroya, NRR
KParczieski, NRR
TSu, NRR
KKniel, NRR
MTaylor, EDO
WDircks, EDO
JRoe, EDO
TRehm, EDO
VStello, EDO

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

Enclosure

MAR 15 1984

MEMORANDUM FOR: Chairman Palladino
Commissioner Gilinsky
Commissioner Roberts
Commissioner Asselstine
Commissioner Bernthal

FROM: William J. Dircks
Executive Director for Operations

SUBJECT: RECOMMENDATION TO DEFER DECISION ON FINAL HYDROGEN CONTROL RULE (SECY-83-357)

The Commission has scheduled an affirmation/discussion and vote on the final Hydrogen Control Rule (SECY-83-357) for March 16, 1984. This memorandum is to inform you of recent test information which I believe warrants further consideration prior to your decision on SECY-83-357.

As part of a cooperative large-scale experimental program at the Nevada Test Station, EPRI has recently completed a series of hydrogen combustion tests in a 52-foot diameter dewar. The objectives of this research program were to confirm test data on hydrogen combustion taken at smaller scale, assess the effects of igniter location on burning and obtain some generic information on the performance and thermal response of selected nuclear plant equipment under a range of hydrogen burn conditions. Preliminary observations (Enclosures 1, 2 and 3) indicate that there may be a potential problem regarding the behavior of electrical cables during a hydrogen burn at a hydrogen concentration of 13 percent. It should also be noted that after entry into TMI-2 following the accident, damage was observed to the electrical cables.

The NRC staff and the contractors are currently in the process of analyzing the data in detail and evaluating the uncertainties and implications of the EPRI tests. This review will include all the equipment identified in Attachment 1 of Enclosure 1. I will continue to keep you advised on this matter as the post-test analyses are conducted and will provide recommendations to you on how and when to proceed with SECY-83-357.

(Signed) William J. Dircks

William J. Dircks
Executive Director
for Operations

Enclosures: See next page

ENCLOSURE "A"

Enclosures:
1. Memo 12/19/83 G. Sliter to Distribution w/2 att
2. Memo 2/17/84 G. Sliter to Distribution w/4 att
3. Memo 2/16/84 R. Curtis to V. Benaroya, W. Butler, K. Kniel & V. Noonan w/ encl

ENCLOSURE "A"

SUMMARY OF PRELIMINARY RESULTS FROM THE EPRI

NEVADA TEST STATION PROGRAM

The objective of the NTS test program was to study hydrogen burning in a relatively large enclosure and to determine effects of the resulting environment on survival of safety related equipment. The tests were performed in a spherical vessel of approximately 52 feet diameter. The vessel was ~~extensively~~ instrumented so that the most important parameters characterizing the hydrogen burn environment could be measured. The equipment exposed to the hydrogen burn was mounted on a special platform located at the center of the sphere. It consisted of several types of pressure transmitters, solenoid valves, motor operated valves, limit switches, fan motors, ignitors, penetrations, RTD's and cables made by several different manufacturers. The tested equipment was operational during the tests with the exception of the cables which were spliced in closed loops and did not carry any current. All the equipment tested was new and unaged. [redacted]

The NTS test program consisted of 40 separate tests. In 24 tests hydrogen was premixed with air and steam before ignition and in 16 tests it was continuously ignited during the test. The test range covered hydrogen concentrations of 5-13 v/o, steam concentration of 5-40 v/o and rates of hydrogen in continuous injection tests of 0.4 - 3.2 kg/minute. Water sprays and mixing fans were used during some of the tests. The equipment was present in 15 tests (both premixed and continuous injection) including the tests with 13 v/o ~~of~~ hydrogen. [redacted] Each piece of equipment was exposed to several tests and was replaced only when it exhibited signs of failure. In many respects the conditions to which the equipment was exposed were more severe than the conditions anticipated to occur in an actual plant during a hydrogen burn. Note that the cable arrangement, in closed loops located near the center of the test sphere, is not typical of that in a nucleer plant where the cables are generally enclosed in conduits or within cable trays. In the NTS tests, equipment was located in a plume of hot gases generated by burning hydrogen and did not have heat sinks which would be present if the equipment were mounted on the walls of the vessel. [redacted] On the other hand the initial temperature of the equipment was lower than it would have been in a post-LOCA condition and because of the scaling effect energy transfer to the equipment was lower. These factors introduced some degree of non-conservatisms to the tests.

W SEVERAL

At present the recorded test data are still being evaluated and a relatively limited amount of mostly qualitative results are available. In general, for the premixed tests with hydrogen concentrations equal to or below 10 v/o most of the equipment did not exhibit significant signs of degradation and with few exceptions did not lose their operational capability. This happened despite the fact that some temperatures exceeded post-LOCA qualification limits. Similarly in the continuous injection tests most equipment performed satisfactorily.

At the hydrogen concentrations higher than 10 v/o the numbers of pieces of equipment exhibiting apparent malfunctions increased due to the effects of hydrogen combustion although a majority of the equipment still performed satisfactorily.

ENCLOSURE "B"

in some cases.

The most severe environments to which the equipment was exposed was a premixed burning with and without water sprays at 13 v/o hydrogen. During these tests operability was significantly degraded. Subsequent examinations revealed ingression of water into instrument boxes in some cases. At this concentration cable insulation exposed to hydrogen flames started to burn leaving its outer surface charred, but not exposing bare conductors. EPRI ascribed many instrument failures to improper installation and pointed out that most of the instruments underwent several tests and so the damage might have been cumulative.

EPRI is planning to issue a preliminary draft of the data evaluation report in the middle of June 1984. This draft will contain only a partial evaluation. The complete evaluation of the data is expected to be accomplished by the end of 1984.

MAY 16 1984

MEMORANDUM FOR: Those on Attached List

FROM: W. S. Farmer, RES

SUBJECT: REPORT OF MAY 10 MEETING WITH EPRI AND ELECTRO-TEST TO REVIEW HYDROGEN BURN CABLE POST-TEST EXAMINATION RESULTS

A meeting was held on May 10 at the Electro-Test Inc. test laboratories in San Ramon, California, to review the results of post-test examination of cable and splice specimens which had been exposed to hydrogen burn tests in the NTS facility. Electro-Test had completed an AC withstand test, IR measurement and DC withstand test for 56 cable specimens, 3 Raychem Class 1E splices and 5 armor splices. Still to be tested are the CONAX and Westinghouse electrical penetrations and the remaining three 48 conductor Firewall III cable specimens (cable type 5). All MV power, LV power and control cable specimens tested to date were found to display adequate insulation resistance and could be expected to function. All four specimens of the Rockbestos 1K8 coaxial cable (cable type 4) failed the electrical tests. In two instances the center conductor was exposed. Individual Rockbestos coaxial cable specimens had seen premixed hydrogen burns at 8, 10 and 13%. Also, one of the four Raychem XLPE coaxial cable specimens (cable type 9) failed the electrical tests. The failure of the cable type 9 coaxial cable specimen was identified as a short between one of the conductors and the braided shield. High voltage coaxial cables are normally used in conjunction with neutron-monitoring instrumentation.

In a telephone conversation on May 16, Dr. George Sliter of EPRI stated that EPRI had contacted Rockbestos and was told that Rockbestos 1K8 coaxial cable had been found in 1980 tests to fail when heated due to differential thermal expansion of the conductor resulting in perforation of the insulation and wire "kinking." These cables are no longer manufactured and are not considered Class 1E. Why MP&L supplied them for the tests is unknown. The only Class 1E coaxial cable failure is, therefore, the one sample of Raychem coaxial cable Type 9C which saw a 13% only hydrogen burn. The other three Raychem coaxial cables passed the electrical tests. The sample which failed will be destructively analyzed to see if the cable had a flaw or was out of specification. It was also reported in this conversation that Electro-Test had completed testing of the remaining three 48 conductor Firewall III Rockbestos cables (Type 5) and all had passed the electrical tests. The 1 conductor that failed out of 48 in the Type 5B cable sample was noted to fail at a relatively low voltage and therefore may be an indication of a void in the insulation as manufactured.

ENCLOSURE "C"

The cable specimens were clustered in seven stacks on the bench for visual inspection (Figure 1). The first stack represented cables removed after test #5 (8% hydrogen premixed burn). The jackets on these cables showed little or no damage. The second stack represented cables removed after test #6 (8% hydrogen burn). Several of the jackets had splits. However, the surface of the jackets exhibited little or no charring. The remaining five stacks represented cables removed after test 8, test 9, test 11, test 14 and test 15 and, accordingly, saw 10% to 13% hydrogen burns. The cable jacket damage increased significantly as the hydrogen volume percentage in the burns was increased from 8 to 13%. It should be noted that the cable jackets provide primarily for mechanical protection. The insulation underneath, which assures the electrical function, appeared unaffected in visual observations.

The 3 Raychem splices that were tested all passed the electrical tests and are reported to have been made up as Class 1E in accordance with Rockbestos instructions. The 5 armor covered splices provided by Duke Power all passed the electrical tests although the armor enclosures contained water and appear to have leaked at the cover gasket. The use of metal enclosures protects the Raychem wire splices inside from the heat and flame of the hydrogen burn and accordingly they were unaffected. However, this does not appear to be a design used extensively at other nuclear facilities.

In conclusion, except for the one Raychem coaxial cable Type 9 and the one conductor in the Firewall III sample Type 5B, all cables passed the post test examination electrical tests and could be expected to function in the event of a hydrogen burn at up to 13% hydrogen in the NTS facility. Those splices which EPRI had identified as Class 1E also passed the post test examination electrical tests and could be expected to function. The cable jackets displayed progressively increased damage as the percent hydrogen in the burns increased from 8 to 13%. However, the jackets provide primarily mechanical protection and appear to have protected the insulation underneath from damage. There does not appear to be any measurable gradation in damage to the insulation with severity of hydrogen burn in the NTS tests. The results of the post-test examination of cables from NTS appear similar to that from TMI-2 where jackets were also damaged but the insulation was undamaged and the cables passed electrical tests. Electro-Test conducted their tests in accordance with good accepted practice.

It was recommended that EPRI consider testing some of the coaxial and triaxial cables by frequency scanning for signals transmission degradation and attenuation. This would detect degradation that might not be apparent from the electrical tests. It would also be of interest to obtain physical property (elongation and tensile strength) measurements of the insulation on some of the TMI cables supplied to NTS in order to compare their property change with that measured at HEDL. A request for cable samples to be sent to HEDL will be made to EPRI.

The cables and splices were all believed by EPRI to be Class 1E. However, documentation was not always provided by the supplying utilities. This documentation should be obtained or the utility be contacted and the categorization of the cables and splices confirmed.

The importance of the coaxial cable failures has been downgraded by the recent (5/16/84) phone conversation with EPRI. It now appears that all Class 1E cables have been shown to pass electrical function tests with the exception of one conductor in one of the four 48 conductor Firewall III cables and one conductor in a Raychem coaxial cable. Further destructive examination of both is to be conducted by EPRI to determine whether the failures were due to flaws in the cable as supplied at the factory.

Original Signed By:

William S. Farmer
Electrical Engineering
Instrumentation & Control Branch
Division of Engineering Technology
Office of Nuclear Regulatory Research

Enclosures:
1. List of Cable Specimens (from EPRI QLR)
2. Cable Quick Look Conditions (from EPRI QLR)
3. Electro-Test Inc. Cable Test Procedures
4. Electro-Test Inc. Cable Electrical Measurements
5. Attendance List, Meeting on May 10, 1984.

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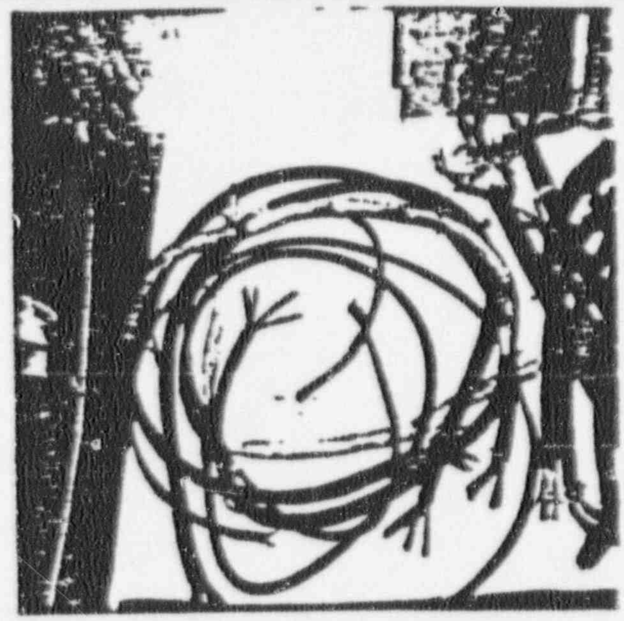


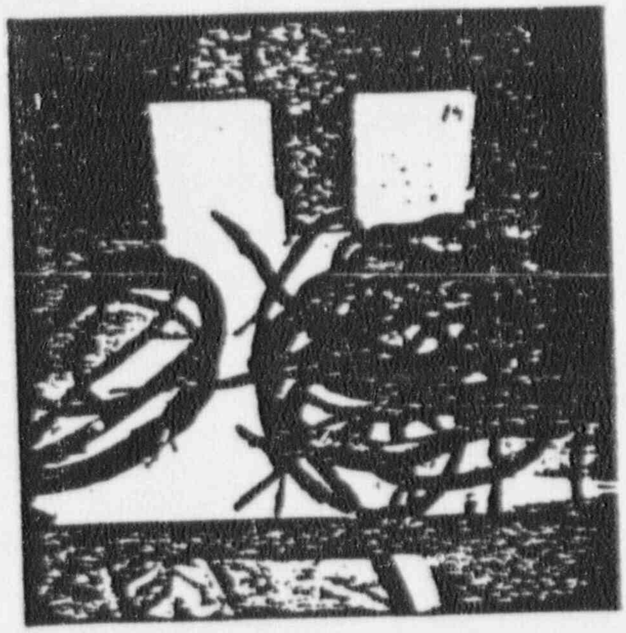
Test #6

Test #5

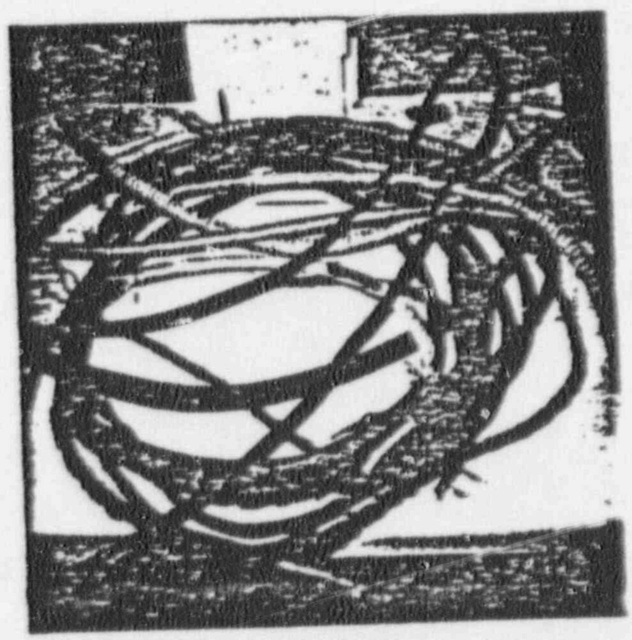
Test #9

Test #[illegible]

F[illegible]



Test #11 Test #14



Test #15

LIST OF CABLE SPECIMENS

CABLE TYPE	CONDUCTORS	GAUGE NO. (AWG)	RATED VOLTAGE (v)	MATERIALS	MANUFACTURER
1.	4/C	12	600	FR III 90°C	Kerite
2.	1/C	4	1000	HTK 90°C	Kerite
3.	4/C	16	600	FR II 90°C	Kerite
4.	1/C (COAX)	21	2100	IKB	Rockbestos
5.	48/C	20	600	Firewall III	Rockbestos
6.	3/C	10	1000	Firewall III	Rockbestos
7.	2/C (STP)	16	600	Dekorad	Samuel Moore
8.	3/C (TRIAX)	16	600	Dekorad	Samuel Moore
9.	2/C (COAX)	18	2000	XLPE	Raychem
10.	4/C	12	1000	Okonite/Okolon/ Okolon	Okonite
11.	3/C	10	1000	Okonite/Okolon/ Okolon	Okonite
12.	2/C	9	600	Silicon Rubber	Kerite
13.	4/C	9	600	Silicon Rubber	Kerite
14.	2/C	12	600	EPR/HYP/HYP	Boston Insulated Wir
15.	5/C	12	600	EPR/HYP/HYP	Boston Insulated Wir
16.	7/C	12	1000	VFR	Okonite
17.	2/C (STP)	16	600	Flameguard FR-EP	Anaconda
18.	3/C	16	600	Silicon Rubber	Boston Insulated Wir
19.	12/C (STP)	16	600	Silicon Rubber	Anaconda
20.	6/C (STP)	16	600	Flameguard FR-EP	Anaconda
21.	1/C (TRIAX)	19	4000	XLPE	Brand-Rex
22.	4/C	16	300	Flame Retardent EP Rubber	Eaton
23.	2/C	16	300	XLPE	Brand-Rex
24.	3/C	12	1000	Flame Retardent EP Rubber	Okonite
25.	3/C	10	2000	Flame Retardent EP Rubber	Anaconda

Not all splices
Class 1E

CABLE TYPE/SPECIMEN	TEST NO. OF LAST EXPOSURE		NUMBER OF PRIOR EXPOSURES	DESCRIPTION OF CONDITION*
1A	5		4	No visible damage
1B	9		3	Char, cracks, one bulge with split through metallic shield braid
1C	14	13%	0	Char, cracks
2A	9		5	Char, cracks, bulges with splits--binder tape visible
2B	8		2	Char, cracks, bulges
2C	14	13%	0	Char, cracks, wrinkles, many small holes--insulation visible
3A	5		4	One split--binder tape visible
3B	6	8%	0	Scorch, one local char, one split--metallic shield braid visible
3C	14	13%	0	Char, cracks
4A	9	1%	5	Char, cracks, wrinkles, metallic shield braid visible many places
4B	6	8%	0	No visible damage except kink with visible conductor--suspect mechanical damage during handling
4C	14	13%	0	Char, wrinkles, few small holes--binder tape visible
4D	15	13%	0	Scorch
5A	9		5	Char, wrinkles, binder tape visible many places
5B	8		2	Char, cracks, wrinkles, small split--binder tape visible
5C	14	13%	0	Char, cracks, wrinkles, binder tape visible many places
5D	15	13%	0	Scorch, cracks, minor local wrinkles
6A	11		7	Char
6B	6	8%	0	Char, cracks, small hole--binder tape visible
6C	14	13%	0	Char, cracks, flakes
6D	15	13%	0	Scorch
7A	5		4	Scorch
7B	6	8%	0	Char, cracks
7C	14	13%	0	Char, wrinkles, flakes
7D	15	13%	0	Scorch, cracks, wrinkles
8A	5		4	Scorch, bulges
8B	6	8%	0	Char, cracks, wrinkles, flakes
8C	14	13%	0	Char, flakes, metallic shield tape visible many places
8D	15	13%	0	Char, wrinkles, flakes
9A	9		5	Char, flakes
9B	6	8%	0	Scorch
9C	14	13%	0	Char, flakes, metallic shield visible many places
9D	15	13%	0	Char

QUICK-LOOK CABLE CONDITION (Cont'd.)

CABLE TYPE/SPECIMEN	TEST NO. OF LAST EXPOSURE		NUMBER OF PRIOR EXPOSURES	DESCRIPTION OF CONDITION*
10A	11		7	Char, cracks
10B	8		2	Char, cracks, bulges
10C	14	13%	[illegible]	Char, cracks, bulges
10D	15	13%	0	Char, cracks
11A	11		7	Scorch, cracks
11B	8		2	Char, cracks, bulges
11C	14	13%	0	Char, cracks, bulges
11D	15	13%	0	Char, cracks
12A	9		8	Char, cracks, flakes
13A	5		4	No visible damage
13B	9		3	Char
14A	11		10	Char, cracks
15A	6		5	Scorch, cracks, one split--binder tape visible
16A	11		10	Char, cracks
17A	9		8	Char, cracks, metallic shield tape visible many places
18A	9		8	Scorch
19A	6		5	Scorch, cracks, many splits--binder tape visible
20A	6		5	Scorch, cracks, metallic shield tape visible one place
21A	15	13%	0	No visible damage to flexible metal armor, interior cables not inspectable
22A	15	13%	0	No visible damage to flexible metal armor, interior cables not inspectable
23A	15	13%	0	No visible damage to flexible metal armor, interior cables not inspectable
24A	15	13%	0	No visible damage to flexible metal armor, interior cables not inspectable
25A	15	13%	0	No visible damage to flexible metal armor, interior cables not inspectable

* Scorch = discoloration or change in texture of surface
Char = soot/ash on surface
Cracks = cracks on surface (depth not discernible)
Flakes = surface pieces of jacket removed, usually not through entire thickness
Bulges = local swelling of jacket
Wrinkles = bulges that form a repeating pattern

• No conductors visible
• Electrical tests
IR Submerged
VW

Electro-Test Inc. Cable Test Procedures

1. AC rated voltage withstand test at 60 cycles by immersing cables in water for 5 minutes.

2. IR test at 300 or 500 volts DC in accordance with IPCEA Standard.

3. DC voltage withstand test in water at 3 times AC test voltage for 5 minutes in accordance with IPCEA S-19-81 Rubber-Insulated Wire Cable.

The ends of the cable insulation were cleaned, a heat shrink sleeve put on and a guard circuit set up. Surface leakage current was negligible in water immersion tests to determine leakage current through the insulation. Multiconductor cable tests consisted of testing a single conductor and grounding the remainder.

Electro-Test Inc. Cable Electrical Measurements

Cables:

1. AC rated voltage test

°	< 1 ma	56 cable specimens
°	> 5 ma	5 cable specimens (4A, 4B, 4C, 4D & 9C)
°	Not yet tested	3 cable specimens (5A, 5C & 5D)

2. Insulation Resistance (IR) at 300 or 500V DC

°	50 K megohm to 100 K megohm	49 cable specimens
°	1 K megohm to 49 K megohm	5 cable specimens
°	75 megohm to 1 K megohm	2 cable specimens
°	< 1 megohm	5 cable specimens (4A, 4B, 4C, 4D & 9C)

3. DC voltage withstand test at 3X voltage

°	< .5 μa	52 cable specimens
°	> .5 μa but < 70 μa	3 cable specimens
°	> 5000 μa	6 cable specimens (4A, 4B, 4C, 4D, 9C & 5B)

Note:

1. Cable Type 5 specimen was under test. One of the 48 wires did not pass the DC voltage withstand test and had a leakage current of > 5000 μa.

Splices:

Worst case values from tests of Raychem Class 1E splices with cables attached:

Cable Type	AC ma	IR megohm	DC μa	Cable Description
6A	.15	100K	.01	3 cond/10/1000V Rockbestos
10A	.15	75K	.04	4 cond/12/1000V Okonite
11A	.15	> 100K	.035	3 cond/10/1000V Okonite

Worst case values for Duke supplied armor protected splices with cables attached:

Cable Type	AC ma	IR megohm	DC μa	Cable Description
21	.12	50K	.02	1 cond/19/1000V Brand Rex
22	.05	> 50K	.02	4 cond/16/300V Eaton
23	.04	> 50K	>.01	2 cond/16/300V Brand Rex
24	.14	> 50K	.01	3 cc[illegible]V Okonite
25	.14	> 50K	.03	3 c[illegible]/1000V [illegible]conda

[illegible]

[illegible]

[illegible]	EPRI	(415) 75[illegible]
[illegible]	[illegible]	[illegible]
[illegible]	[illegible]	415) [illegible]
Dick Miller	Westinghouse	412-378-5217
Phil Cannon	HEDL	509-376-3653
Bill [illegible]	NRC	301-443-7946
[illegible] King	[illegible]	[illegible]
Archie Tewarson	FMRC	617-762-4300
JOHN J. HAUGH	EPRI	415/855-2768
JOHN WANLESS	NUS	415/854-8900
JEAN-PIERRE WOLFF	ELECTRO-TEST INC.	415 [illegible]-5666
[illegible]	ELECTRO-TEST INC.	415 [illegible]
JOHN MOORE	ELECTRO-TEST INC.	415 [illegible] 5666
DAVID GSCHWEND	Electro-Test Inc.	415 820-5666

THE PRODUCTION OF H_2 IN LWR SYSTEMS DURING SEVERE ACCIDENTS

The early production of H_2 in LWR systems during a severe accident is mainly caused by the oxidation of the Zircaloy cladding by steam. This reaction produces one H_2 molecule for every steam molecule reacted. The geometry of an undamaged (i.e., unmelted core) is very susceptible to oxidation by steam at temperatures above 2200°F (the current Appendix K limitation for LOCA events). The very high surface to volume ratio of the Zircaloy cladding allows sufficient contact area for the oxidation to occur.

When an LWR undergoes a severe accident leading to loss of coolant, it eventually reaches the stage at which the primary water is gradually boiled away leaving the core uncovered. For rapid blowdown events such as a large-break LOCA, accumulator injection will refill the core. If the low pressure injection system then fails, the core water will boil off slowly at ambient pressure. Boilaway times from the top of the core to the bottom are of the order of 20-40 minutes depending on the exact sequence if no additional water is injected into the core by operator action (Note that at TMI-2 water was injected periodically by operator action.)

State of knowledge calculations and direct experimental data (PBF Severe Fuel Damage Series I, Ex-Pile experiments at KfK in the FRG*) show that for an unattenuated boilaway only 15-25% of the Zircaloy cladding will be fully oxidized prior to gross melting of the core (i.e., when temperatures in excess of 5000°F are reached, and failure of the lower core support plate and lower plenum occur because of the high heating rate incurred.) After gross melting and core collapse into the reactor building lower cavity, the remaining unoxidized Zircaloy may undergo further oxidation depending upon

*Kernforschungszentrum-Karlsruhe in the Federal Republic of Germany

Enclosure "D"

the availability of oxidant (steam) and the ability of the Zircaloy (present now with a considerably lower surface to volume ratio within a large mass of molten UO_2, ZrO_2, steel, etc.) to come in contact with the steam. Very little information is available on the oxidation rate in such a system but experiments are planned to obtain the needed data.

For accident sequences where an unattenuated boiloff is prevented by periodic water injections (bleed and feed), somwhat more oxidation of the Zircaloy can be achieved prior to or without gross melting of the core. A situation similar to this occurred at TMI-2 where ~45-50% of the cladding was oxidized. However, there is a limit to the total oxidation possible for this type of sequence also. Maximum oxidation (i.e., maximum H_2 production) will occur if the core is boiled down to certain water levels and held there by feed and bleed operations. However, the water level has to be low enough for considerable oxidation but high enough to provide a sufficient steaming rate to preclude gross melting of the core. Calculations included by the Rogovin Study Group in their report on Three-Mile Island (Volume 2, Part 2, Appendix II.8) illustrate the previous point. Figures 1A, 1B, & 1C show core temperatures achieved for uncoveries of 7-feet, 8-feet, and 9-feet respectively for the TMI-2 core. The individual curves on each plot indicate the axial location of the position from the <u>top</u> of the core. That is, 0 = top of core, 1 = one foot down from top, 2 = two feet down from top, etc. All calculations assumed a boildown to the indicated level in

20 minutes and holding at that level (presumably by bleed & feed) for an additional 60 minutes. To interpret these curves in terms of the production one must understand that very rapid oxidation of Zircaloy will not occur until the Zircaloy temperature exceeds 2200°F. Using the kinetic equation for oxidation of Zircaloy given in NUREG/CR-0497, TREE-1280, Revision 2, one can calculate the percentage of cladding oxidized in one hour at various temperatures. Table 1 shows the results at three temperatures: 1500°F, 1830°F and 2200°F. Note that below 1500° only 2% oxidation will occur in one hour. Since the kinetics are a function of the square root of the time, only 2.8% oxidation will occur after 2 hours, etc. Therefore, for those regions of the core below say 1500°F, it can be safely assumed that very little H_2 will be produced (~less than 3%).

Figure 1A shows, therefore, that for seven feet of the core uncovered for one-hour ~4 feet of the cladding is potentially totally oxidizable. That is, a total H_2 production of only 4/12 x 100 = 33%. For eight feet of the core uncovered (Figure 1B), 5 feet of the core could be totally oxidized leading to a total possible H_2 production from the clad of 5/12 x 100 = 42%. This latter case is considered to be close to what actually happened at TMI-2. Note that for the above two situations no part of the core exceeded 4400°F which is above the cladding melting of point but below the fuel (UO_2) melting point of just over 5000°F. Thus, no gross melting of the core would have taken place except for some dissolution of UO_2 into the liquid Zircaloy (in the upper two feet only).

Figure 1C shows the results for a nine-foot uncovery and hold. In this case 7/12 x 100 = 60% H_2 would be produced. However, note that 3.5 feet of the core will now reach the UO_2 melting point and core collapse and gross melting will begin. This large mass may cause failure of the lower plate and lower plenum and lead to a "core on the floor" event. That is, the core would not be recoverable. Note also that even this very severe event oxidized only ~60% of the cladding prior to core collapse.

Therefore, it seems highly unlikely that an event can be constructed to result in oxidation of more than about 50-60% of the cladding and still maintain the core in a recoverable state (i.e., no gross melting and relocation). Experimental data to confirm such analyses will be performed on full-length fuel bundles by NRC in the NRU reactor in Canada over the next year. Follow-on tests will be needed to confirm the calculations for various uncovery levels.

Table 1. Percent Cladding Oxidized at Various Temperatures in One Hour

Temperature (^oF)		Present Clad Oxidized
2200	(1200^oC)	20.5
1830	(1000^oC)	7.8
1500	(800^oC)	2.1

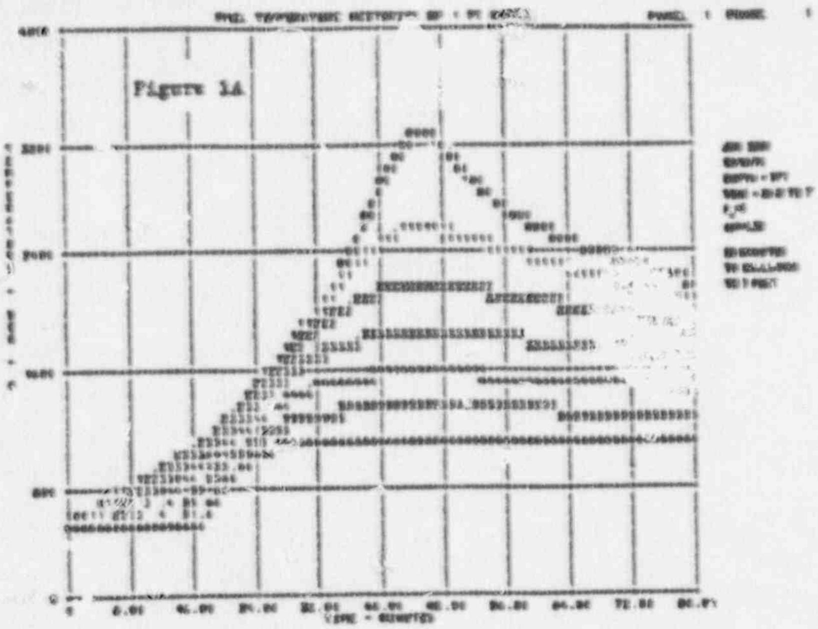


Figure 1A

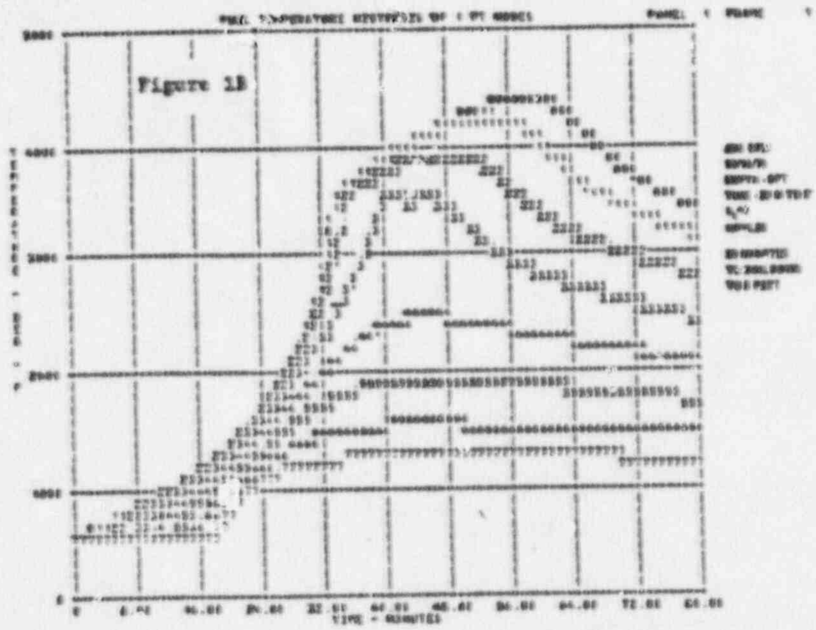


Figure 1B

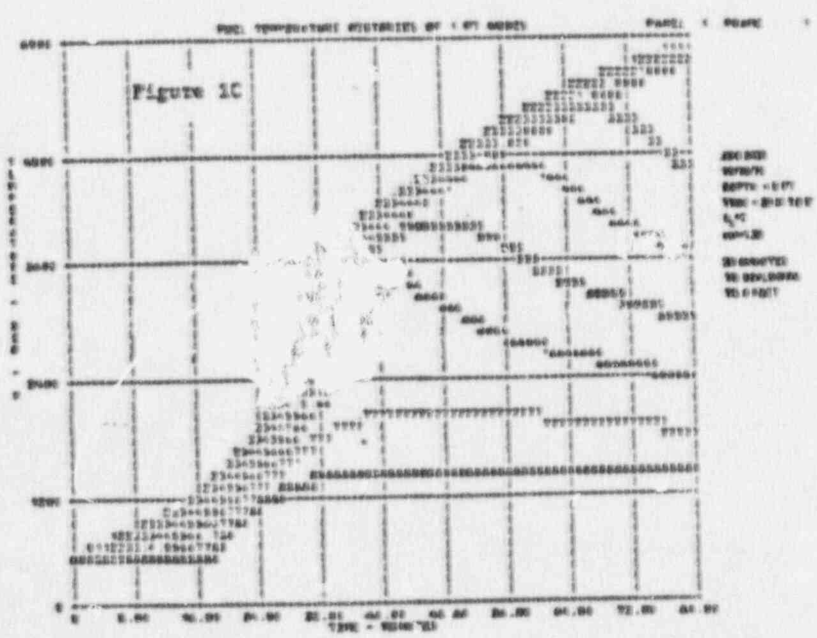


Figure 1C

Accident Likelihood and Potential for "Arrested" Sequences in PWRs with Large, Dry Containments

Since the time of the original staff proposals on interim hydrogen rulemaking (SECY 81-245A), a considerable amount of analysis has been performed with respect to the likelihood of severe accident sequences in PWRs with large dry containments. A number of PRAs have been performed under the sponsorship of both the NRC and utilities, and have identified those sequences which contribute most importantly to the respective estimates of the frequency of core melting. Such information has been catalogued and studied as one task of the RES-sponsored Accident Sequence Evaluation Program (ASEP). Based on this catalogue, it has been determined that transient-initiated accidents and small LOCAs are, in general, the most important types of sequences in most PWRs. More specifically, ASEP identified five accident sequences which are generally important to the characterization of the core melt frequency in PWRs. These sequences are:

- Transient without early emergency core cooling
- Transient-induced LOCA without early emergency core cooling
- Small pipe break LOCA without early emergency core cooling
- Transient without reactor shutdown (ATWS)
- Transient without early emergency core cooling and without containment heat removal.

ENCLOSURE "E"

An important characteristic of these sequences (except ATWS) is that the loss of water from the reactor coolant system is rather gradual, and thus the processes of core uncovery, core damage, and gross core melting can be protracted. For this reason, there can be a significant potential for operator intervention to restore cooling of the fuel. In general, this recovery potential has not been included in published PRAs. One exception to this is the incorporation of the recovery of offsite electric power during a station blackout accident sequence. In this sequence, a recovery probability (as a function of time) based on historical data is typically included.

The potential benefit of accounting for operator recovery actions has been evaluated as part of the NRC-sponsored IREP PRA of the Arkansas Nuclear One (ANO) Unit 1 plant. In this study, faults which by themselves or in combination with other faults could lead to failures of important systems were categorized as recoverable or non-recoverable (e.g., an inadvertently closed valve was recoverable, a plugged valve was not). For each recoverable fault, a recovery location was identified and a critical time developed, based on how quickly the specific associated function had to be restored. The latter time was based on the onset of an "unacceptable" core condition. For the IREP study, the unacceptable condition was specified as core uncovery, which for transient events and small LOCAs occurred in roughly forty minutes. Using this data and general data on the probabilities of human restoration actions as functions of time, a recovery probability was developed for each recoverable fault.

Since the publication of the ANO IREP, this analysis has been extended to consider restoration times correlated to an unacceptable core condition defined as onset of core melting, rather than core uncovery. In effect, this extended the recovery time for small LOCAs and transients (including loss and subsequent recovery of offsite power) from roughly forty minutes to a time on the order of seventy minutes. Table 1 displays the results of these analyses.

In addition to these assessments of recovery potential, the ANO IREP also identified two accident sequences in which it was possible that the operable core cooling equipment might be adequate to prevent gross core melting, even though it would apparently not prevent core uncovery. In Table 1, the last column ("alternative success criterion") includes this possible effect for the two specific sequences.

The table 1 results suggest several conclusions. With respect to operator recovery actions, it appears that, for this ANO case study:

- operator recovery actions <u>before</u> core uncovery can have a significant effect on estimates of core melt frequency; and

- if recovery actions have not occurred by the time of core uncovery, it is unlikely that such actions will be taken before the onset of gross core melting.

The second conclusion results from two factors. First, many recoverable faults can be rectified very quickly, and, as such, are accounted for in the pre-core uncovery phase (thus the first conclusion). As time proceeds, however, the incremental changes in probability due to recovery actions decrease rapidly. That is, the recovery model corresponds to the intuitive concept that if the operators have been unable to correctly diagnose and account for initial faults by the time the core is first uncovered, then the likelihood of such actions by the time of gross core melting is relatively low. In addition, as such credit for recovery actions decreases the effect of <u>recoverable</u> faults, <u>non-recoverable</u> faults become increasingly more important and finally prevent any further gain. Thus, as Table 1 shows, the only incremental reduction in probability occurs in the station blackout sequence (T(LOP) $\overline{\text{AFW}}$ $\overline{\text{ECC}}$ $\overline{\text{SPRAYS RBC}}$), due to the somewhat greater likelihood of offsite power restoration prior to core melting. The inclusion of the probability reduction due to an alternative success criterion for two sequences has a more pronounced effect, but the overall reduction still remains rather small.

CONTRARY TO ARGUMENT FOR AC IGNITER

In summary, for this case study, it can be concluded that recovery actions are a significant factor in preventing severe core damage. However, such actions are most likely to occur before any damage has occurred (i.e., before core uncovery). If the accident has progressed to the point of initial core uncovery, then, under the assumptions used for this case study, it has relatively little chance of being arrested before gross core melting.

TABLE 1

EFFECT OF RECOVERY ASSUMPTIONS ON DOMINANT ACCIDENT SEQUENCE FREQUENCIES

FREQUENCY (PER R. YR.)

SEQUENCE	NO RECOVERY	RECOVERY< CORE UNCOVERED	RECOVERY< CORE MELT	ALTERNATIVE SUCCESS CRITERION
SLOCA (1.2) ECC	2E-5	3E-6	3E-6	
SLOCA (1.2) ECC SPRAYS	2E-5	4E-6	4E-6	
SLOCA (1.7) ECCR	3E-5	1E-6	1E-6	
SLOCA (4) ECCR	4E-5	1E-6	1E-6	
T(A3) S/RV AFW ECC	7E-6	3E-6	3E-6	1 E-7
T(D01) S/RV AFW ECC	2E-5	4E-6	4E-6	1 E-7
T(LOP) AFW ECC SPRAYS RBC	4E-5	1E-5	6E-6	
T(F1A) RPS ECC	3E-6	3E-6	3E-6	
T(A3) AFW ECC	6E-6	1E-6	1E-6	
T(A3) AFW ECC SPRAYS	3E-6	1E-6	1E-6	
T(D01) AFW ECC	2E-5	2E-6	2E-6	
T(D01) AFW ECC SPRAYS	1E-5	2E-6	2E-6	
T(D01) AFW ECC SPRAYS RBC	5E-5	3E-6	3E-6	
T(D02) AFW ECC SPRAYS RBC	6E-6	3E-6	3E-6	
ALL DOMINANT SEQUENCES	3E-4	4.1E-5	3.7E-5	3 E-5

TABLE 1 (continued)

TERMS

SLOCA (1.2)	Small LOCA (.4" - 1.2")
SLOCA (1.7)	Small LOCA (1.2" - 1.7")
SLOCA (4)	Small LOCA (1.7" - 4")
T(LOP)	Transient initiated by loss of offsite power
T(A3)	Transient initiated by failure of 4160 VAC bus A3
T(D01)	Transient initiated by failure of 125 VDC bus D01
T(D02)	Transient initiated by failure of 125 VDC bus D02
T(FIA)	Transient with all front-line systems (ECC, AFW, etc.) initially available
$\overline{\text{ECC}}$	Failure of emergency core cooling system in injection mode
$\overline{\text{ECCR}}$	Failure of emergency core cooling system in recirculation mode
$\overline{\text{SPRAYS}}$	Failure of containment spray system
$\overline{\text{S/RV}}$	Stuck-open safety/relief valve
$\overline{\text{AFW}}$	Failure of auxiliary feedwater system
$\overline{\text{RBC}}$	Failure of containment fan cooling system

ENCLOSURE 5

COMMENTS
TO MORT FLEISHMAN
12/7/84

COMMISSION BRIEFING

ON

HYDROGEN CONTROL FOR MARK III BWRs

AND ICE CONDENSER PWRs

DECEMBER 10, 1984

MALCOLM L. [illegible]
443-7923
DECEMBER 10, 1984

STATUS OF HYDROGEN RULE

EXTENDED COMMENT PERIOD EXPIRED APRIL 8, 1982

MAJOR COMMENTS RECEIVED

- DEFER TO SEVERE ACCIDENT RULEMAKING
- 75% METAL-WATER REACTION IS TOO HIGH
- REQUIREMENT FOR CONTAINMENT INTEGRITY TOO RESTRICTIVE
- SHOULD NOT HAVE TO ANALYZE ALTERNATIVE SYSTEMS
- COLD SHUTDOWN IS INCONSISTENT WITH LIC. BASIS
- SCHEDULES ARE UNREALISTIC

• E.Q. VS SURVIVABILITY

MEETING WITH COMMISSIONERS	11/09/83
RESPONDED TO COMMISSIONERS COMMENTS	12/28/83
INITIAL NTS TEST RESULTS	JAN. 83
REQUESTED COMMISSIONERS DELAY DECISION	03/15/84
POST TEST EXAMINATIONS COMPLETED	JULY 84 ?

POST TEST EXAMINATIONS OF CABLE

POST TEST EXAMINATIONS OF EQUIPMENT

MALCOLM L. ERNST
443-7923
DECEMBER 10, 1984

BACKGROUND

FINAL RULE - PENDING CP/ML — JANUARY 15, 1982

- o H_2 CONTROL - 100% METAL-WATER REACTION
- o CONTAINMENT PRESSURE
- o SYSTEM AND COMPONENT FUNCTION
- o OTHER DEGRADED CORE ISSUES

FINAL RULE - MARK I AND II — DECEMBER 2, 1981

- o INERT CONTAINMENT
- o RECOMBINER CAPABILITY
- o HIGHPOINT VENTS

PROPOSED RULE - MARK III AND ICE CONDENSERS — DECEMBER 23, 1981 (COMMENT PERIOD EXTENDED ON FEBRUARY 25, 1982)

- o H_2CONTROL - 75% METAL-WATER REACTION
- o EQUIPMENT QUALIFICATION
- o ANALYSES

MALCOLM L. ERNST
443-7923
DECEMBER 10, 1984

PRINCIPAL FEATURES OF RULE

APPLIES ONLY TO MARK III AND ICE CONDENSER PLANTS

- o REQUIRES HYDROGEN CONTROL SYSTEMS
  - 75% FUEL CLADDING - WATER REACTION
  - NO LOSS OF CONTAINMENT INTEGRITY
- o FUNCTIONING OF SYSTEMS AND COMPONENTS DURING AND AFTER HYDROGEN BURN
  - NEEDED TO ESTABLISH AND MAINTAIN SAFE SHUTDOWN AND CONTAINMENT INTEGRITY
  - LOCAL DETONATIONS INCLUDED UNLESS UNLIKELY TO OCCUR
- o SUPPORTING ANALYSES
- o IMPLEMENTATION
  - SUBMIT IMPLEMENTATION SCHEDULE WITHIN 6 MONTHS
  - MUTUAL AGREEMENT - LICENSEE AND NRC STAFF

MALCOLM L. ERNST
443-7923
DECEMBER 10, 1984

SUMMARY OF NTS TESTS

PURPOSE

DEVELOP DATA BASE FOR VALIDATION OF ANALYTICAL MODELS &

- ~~CONFIRM~~ VERIFY SMALLER SCALE TEST DATA ON H_2 COMBUSTION
- ~~ASSESS EFFECTS OF IGNITER LOCATION ON BURNING~~ { VERIFY EFFECTIVENESS OF DELIBERATE IGNITION APPROACH TO H_2 CONTROL
- ~~OBTAIN GENERIC INFORMATION ON EQUIPMENT RESPONSE UNDER HYDROGEN BURN CONDITIONS~~ MEASURE EQUIP RESPONSE DURING & AFTER H_2 COMBUSTION EVENTS

DESCRIPTION

- HYDROGEN COMBUSTION IN 52' D SPHERICAL VESSEL
- 5 - 13 V/O H_2 (PREMIXED)
- 5 - 40 V/O STEAM
- 0.4 - 3.2 KG/MIN H_2 (CONTINUOUS INJECTION)
- EQUIPMENT EXPOSED TO MULTIPLE BURNS
  - PRESSURE TRANSMITTERS
  - VALVES
  - MOTORS
  - CABLES
  - PENETRATION ASSEMBLIES, ETC.
- EQUIPMENT WAS OPERATIONAL EXCEPT FOR CABLES (INCLUDING ELECTRICAL PENETRATIONS)

[redacted]

RESULTS

- CABLES EXTERNALLY DAMAGED DURING HIGHER CONCENTRATION TESTS
  - POST TEST EXAMINATION REVEALED ALL BUT TWO CABLES WERE FUNCTIONAL
- ELECTRICAL PENETRATION ASSEMBLIES HAD 38 OF 52 LEADS PASS POST TEST EXAMINATIONS
  - NOT KNOWN IF FAILURES DUE TO SINGLE BURN OR CUMMULATIVE EFFECT OF MULTIPLE BURNS

MALCOLM L. ERNST
443-7923
DECEMBER 10, 1984

STAFF CONCLUSIONS CONCERNING NTS TEST RESULTS

- TESTS NOT DIRECT DEMONSTRATION OF EQUIPMENT SURVIVABILITY DUE TO DIFFERENCES IN SCALE AND CONDITIONS
  - ~~1/20 SCALE TEST~~ 75,000 ft^3
  - LOCATIONS OF EQUIPMENT
  - ~~HEAT LOADS WERE LOWER~~ (NO BASIS AT PRESENT TIME)
  - EQUIPMENT NOT AGED
  - CABLES NOT ENCLOSED IN CONDUITS (EXCEPTION OF ARMORED CABLE)
  - CABLES NOT ENERGIZED
  - ~~NO HEAT SINKS~~
  - WATER SPRAYS NOT AS DENSE
- SOME TEST CONDITIONS CONSERVATIVE, SOME NONCONSERVATIVE

MALCOLM L. ERNST
443-7923
DECEMBER 10, 1984

RESEARCH ACTIVITIES UNDERWAY

o HCOG 1/4 SCALE DIFFISION FLAME TESTS FOR MARK III — EARLY 1985

o SNL WORK

- SMALL SCALE TESTS (?) — EARLY 1986

- ? EXPERIMENTAL AND ANALYTICAL STUDIES BASED ON NTS GENERATED DATA TO DETERMINE STRESS ON EQUIPMENT — EARLY 1986

- LOCAL DETONATION ANALYSES FOR LARGE, DRY PWRs — LATE 1985

- DIFFUSION FLAME MODEL DEVELOPMENT (?) — MID 1985

MALCOLM L. ERNST
443-7923
DECEMBER 10, 1984

STAFF RECOMMENDATIONS

- APPROVE H_2 CONTROL RULE FOR MARK III BWRs AND ICE CONDENSER PWRs
- DEFER RULEMAKING FOR LARGE, DRY PWRs UNTIL COMPLETION OF RESEARCH EFFORT
- STAFF WILL RECOMMEND WHETHER RULEMAKING FOR LARGE, DRY PWRs IS WARRANTED IN MID 1986

MALCOLM L. ERNST
443-7923
DECEMBER 10, 1984

CONCLUSIONS

- o RATIONALE FOR THE STAFF RECOMMENDATIONS IN SECY 83-357
  - CODIFY PAST COMMISSION PRACTICE (SEQUOYAH - McGUIRE)
  - TMI EXPERIENCE
  - GREATER INHERENT CAPABILITIES OF LARGE-DRY CONTAINMENTS
  - JUDGMENT THAT CONDITIONS DURING BURN WOULD BE NO WORSE THAN CONDITIONS DURING LOCA
- o NTS TESTS
  - ~~CABLE BURNS CAST DOUBT REGARDING EQUIPMENT QUALIFICATION JUDGMENT~~
- o ANALYSIS OF NTS TEST SHOWS:
  - ~~ALL~~ MOST EQUIPMENT FUNCTIONAL, SOME EQUIP. CABLES AND ELECTRICAL PENETRATIONS NOT FUNCTIONAL
  - TEST ~~DID NOT SIMULATE REAL ACCIDENT~~ CONDITIONS
- O STAFF BELIEVES THAT NTS EXPERIMENTS DO NOT INVALIDATE RECOMMENDATIONS MADE IN SECY 83-357

MALCOLM L. ERNST
443-7923
DECEMBER 10, 1984

STAFF RECOMMENDATIONS

- o APPROVE H_2 CONTROL RUL[illegible] [illegible]R MARK III BWRs AND ICE CONDENSER PWRs
- o STAFF WILL RECOMMEND WHETHER RULEMAKING FOR LARGE, DRY PWRs IS WARRANTED IN MID 1986

MALCOLM L. ERNST
443-7923
DECEMBER 10, 1984

Central File

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

July 3, 1985

MEMORANDUM FOR: Cecil O. Thomas, Chief
Standardization and Special Projects Branch
Division of Licensing

FROM: Dennis M. Crutchfield, Assistant Director
for Safety Assessment
Division of Licensing

SUBJECT: PROOF AND REVIEW OF THE CESSAR SYSTEM 80 TECHNICAL SPECIFICATIONS

The enclosed technical specifications (TS) for CESSAR System 80 Amendment 10 (CESSAR 80) are being forwarded to you to be sent to Combustion Engineering (CE) for proof and review. The letter transferring the enclosed technical specifications to CE should include the following: a request to return their comments by July 17, 1985; a request to justify by discussion any changes requested to the enclosed TS and a request to provide the numbers in the CESSAR FSAR that should be in CESSAR 80 TS Tables 2.2-1, 3.3-2, 3.3-4 and 3.3-5. CE should be informed that the enclosed CESSAR 80 TS are based on its submittal and the staff's comparison of the NSSS part of the PVNGS-1 TS to the Combustion Engineering STS Revision 3 as to whether differences were systems 80 differences or plant specific differences

The attachment provides an NSSS TS for the CESSAR System 80 standardized plant design. Because Palo Verde Nuclear Generating Station Unit One (PVNGS-1) is the first CESSAR 80 plant, the CESSAR 80 (as opposed to BOP) part of the PVNGS-1 TS has been selected as a straw-man for an NSSS Standardized Technical Specification (STS) for CESSAR 80. Therefore, the attached proof and review CESSAR 80 TS for CE's review and comments are a marked-up copy of the NSSS part of the PVNGS-1 TS. Plant specific aspects of CESSAR 80 design in the enclosed marked-up PVNGS-1 TS have either been annotated with references to the applicant's Safety Analysis Report (SAR) or deleted from the TSs.

Mr. Jack Donohew, of ORB#5 will be available during the proof and review period to answer any questions which arise. He is located in Room 310, Phillips, and his telephone extension is 49-27176.

Even if CE has no comments and is in agreement with the technical specifications content, it is requested that a written response from CE to that effect be provided by the above specified date.

Dennis M. Crutchfield

Dennis M. Crutchfield, Assistant Director
for Safety Assessment
Division of Licensing

Enclosure:
CESSAR System 80 Amendment 10 Proof
and Review Technical Specifications

cc: w/o enclosure
H. Thompson
J. Zwolinski
S. Brown
C. Moon
P. Moriette

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Dennis M. Crutchfield, Assistant Director
for Safety Assessment
Division of Licensing

Enclosure:
CESSAR System 80 Amendment 10 Proof
and Review Technical Specifications

cc: w/o enclosure
H. Thompson
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S. Brown
C. Moon
P. Moriette

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7/3/85	7/3/85	7/3/85

June 29, 1985

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(NSSS) TECHNICAL SPECIFICATIONS

CESSAR SYSTEM 80

~~PALO VERDE NUCLEAR GENERATING STATION~~

UNIT NO. ~~[illegible]~~

DOCKET NO. ~~[illegible]~~ 50-470

DEFINITIONS

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SECTION | PAGE

1.0 DEFINITIONS

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INDEX

DEFINITIONS

SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

BASES

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LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

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LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

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LIMITING CONDITION FOR OPERATION AND SURVEILLANCE REQUIREMENTS

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LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

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SECTION — PAGE

80

Insert 2

Insert 1

INSERT 2

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LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

SECTION PAGE

Stet

CESSAR80-NSSS-STS

~~Amendment Number 8~~
~~February 27, 1984~~

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INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

~~Amendment Number 9~~
~~February 27, 1984~~

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LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

BASES

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ADMINISTRATIVE CONTROLS

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PROOF & REVIEW COPY

LIST OF FIGURES

PROOF & REVIEW COPY

FUEL CYCLE FOR MODES 3,4, AND 5

LIST OF TABLES

PROOF & REVIEW COPY

DURING TWO LOOP OPERATION WITH FOUR PUMPS OPERATING

LIST OF TABLES

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SECTION 1.0

DEFINITIONS

CESSAR-80-NSSS-STS

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The defined terms of this section appear in capitalized type and are applicable throughout these Technical Specifications.

ACTION

1.1 ACTION shall be that part of a specification which prescribes remedial measures required under designated conditions.

AXIAL SHAPE INDEX

1.2 The AXIAL SHAPE INDEX shall be the power generated in the lower half of the core less the power generated in the upper half of the core divided by the sum of these powers.

AZIMUTHAL POWER TILT - T_q

1.3 AZIMUTHAL POWER TILT shall be the power asymmetry between azimuthally symmetric fuel assemblies.

CHANNEL CALIBRATION

1.4 A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel output such that it responds with the necessary range and accuracy to known values of the parameter which the channel monitors. The CHANNEL CALIBRATION shall encompass the entire channel including the sensor and alarm and/or trip functions, and shall include the CHANNEL FUNCTIONAL TEST. The CHANNEL CALIBRATION may be performed by any series of sequential, overlapping, or total channel steps such that the entire channel is calibrated.

CHANNEL CHECK

1.5 A CHANNEL CHECK shall be the qualitative assessment of channel behavior during operation by observation. This determination shall include, where possible, comparison of the channel indication and/or status with other indications and/or status derived from independent instrument channels measuring the same parameter.

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See Applicant's SAR.

CHANNEL FUNCTIONAL TEST

1.6 A CHANNEL FUNCTIONAL TEST shall be:

a. Analog channels - the injection of a simulated signal into the channel as close to the sensor as practicable to verify OPERABILITY including alarm and/or trip functions.
b. Bistable channels - the injection of a simulated signal into the sensor to verify OPERABILITY including alarm and/or trip functions.
c. Digital computer channels - the exercising of the digital computer hardware using diagnostic programs and the injection of simulated process data into the channel to verify OPERABILITY including alarm and/or trip functions.
d. ~~Radiological effluent process monitoring channels - the CHANNEL FUNCTIONAL TEST may be performed by any series of sequential, overlapping, or total channel steps such that the entire channel is functionally tested.~~

The CHANNEL FUNCTIONAL TEST shall include adjustment, as necessary, of the alarm, interlock and/or trip setpoints such that the setpoints are within the required range and accuracy.

CONTAINMENT INTEGRITY

1.7 CONTAINMENT INTEGRITY shall exist when:

a. All penetrations required to be closed during accident conditions are either:
   1. Capable of being closed by an OPERABLE containment automatic isolation valve system, or
   2. Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions, except as provided in Table 3.6-1 of Specification 3.6.3.
b. All equipment hatches are closed and sealed,
c. Each air lock is in compliance with the requirements of Specification 3.6.1.3,
d. The containment leakage rates are within the limits of Specification 3.6.1.2, and
e. The sealing mechanism associated with each penetration (e.g., welds, bellows or O-rings) is OPERABLE.

CONTROLLED LEAKAGE

1.8 Not Applicable.

CORE ALTERATION

1.9 CORE ALTERATION shall be the movement or manipulation of any component within the reactor pressure vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATION shall not preclude completion of movement of a component to a safe conservative position.

DOSE EQUIVALENT I-131

1.10 DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134 and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites."

Ē - AVERAGE DISINTEGRATION ENERGY

1.11 Ē shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than iodines, with half-lives greater than 15 minutes, making up at least 95% of the total noniodine activity in the coolant. X

ENGINEERED SAFETY FEATURES RESPONSE TIME

1.12 The ENGINEERED SAFETY FEATURES RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ESF actuation setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays where applicable.

FREQUENCY NOTATION

1.13 The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1.1.

GASEOUS RADWASTE SYSTEM ← BOP Definition

~~1.14 A GASEOUS RADWASTE SYSTEM shall be any system designed and installed to reduce radioactive gaseous effluents by collecting primary coolant system offgases from the primary system and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.~~

IDENTIFIED LEAKAGE

1.15 IDENTIFIED LEAKAGE shall be:

a. Leakage into closed systems, other than reactor coolant pump controlled bleed-off flow, such as pump seal or valve packing leaks that are captured and conducted to a sump or collecting tank, or

b. Leakage into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be PRESSURE BOUNDARY LEAKAGE, or

c. Reactor Coolant System leakage through a steam generator to the secondary system.

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MEMBER(S) OF THE PUBLIC

BOP Definition

~~1.16 MEMBER(S) OF THE PUBLIC shall include all persons who are not occupationally associated with the plant. This category does not include employees of the licensee, its contractors, or vendors. Also excluded from this category are persons who enter the site to service equipment or to make deliveries. This category does include persons who use portions of the site for recreational, occupational, or other purposes not associated with the plant.~~

OFFSITE DOSE CALCULATION MANUAL (ODCM)

BOP Definition

~~1.17 The OFFSITE DOSE CALCULATION MANUAL shall contain the current methodology and parameters used in the calculation of offsite doses due to radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm/trip setpoints, and in the conduct of the environmental radiological monitoring program.~~

OPERABLE - OPERABILITY

1.18 A system, subsystem, train, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s), and when all necessary attendant instrumentation, controls, electrical power, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its function(s) are also capable of performing their related support function(s).

OPERATIONAL MODE - MODE

1.19 An OPERATIONAL MODE (i.e. MODE) shall correspond to any one inclusive combination of core reactivity condition, power level, and cold leg reactor coolant temperature specified in Table 1.2.

PHYSICS TESTS

1.20 PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation and (1) described in Chapter 14.0 of the FSAR, (2) authorized under the provisions of 10 CFR 50.59, or (3) otherwise approved by the Commission.

PLANAR RADIAL PEAKING FACTOR - F_{xy}

1.21 The PLANAR RADIAL PEAKING FACTOR is the ratio of the peak to plane average power density of the individual fuel rods in a given horizontal plane, excluding the effects of azimuthal tilt.

PRESSURE BOUNDARY LEAKAGE

1.22 PRESSURE BOUNDARY LEAKAGE shall be leakage (except steam generator tube leakage) through a nonisolable fault in a Reactor Coolant System component body, pipe wall, or vessel wall.

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BOP Definition

PROCESS CONTROL PROGRAM (PCP)

~~1.23 The PROCESS CONTROL PROGRAM shall contain the provisions to assure that the SOLIDIFICATION of wet radioactive wastes results in a waste form with properties that meet the requirements of 10 CFR Part 61 and of low level radioactive waste disposal sites. The PCP shall identify process parameters influencing SOLIDIFICATION such as pH, oil content, H_2O content, solids content, ratio of solidification agent to waste and/or necessary additives for each type of anticipated waste, and the acceptable boundary conditions for the process parameters shall be identified for each waste type, based on laboratory scale and full-scale testing or experience. The PCP shall also include an identification of conditions that must be satisfied, based on full-scale testing, to assure that dewatering of bead resins, powdered resins, and filter sludges will result in volumes of free water, at the time of disposal, within the limits of 10 CFR Part 61 and of low level radioactive waste disposal sites.~~

BOP Definition

PURGE - PURGING

~~1.24 PURGE or PURGING shall be the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration, or other operating condition, in such a manner that replacement air or gas is required to purify the confinement.~~

RATED THERMAL POWER

1.25 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 3800 MWt. X

REACTOR TRIP SYSTEM RESPONSE TIME

1.26 The REACTOR TRIP SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until electrical power is interrupted to the CEA drive mechanism.

REPORTABLE EVENT

1.27 A REPORTABLE EVENT shall be any of those conditions specified in Section 50.73 to 10 CFR Part 50.

INSERT

SHUTDOWN MARGIN

1.28 SHUTDOWN MARGIN shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming:

a. No change in part-length control element assembly position, and

b. All full-length control element assemblies (shutdown and regulating) are fully inserted except for the single assembly of highest reactivity worth which is assumed to be fully withdrawn.

Insert

BOP Definition

SHIELD BUILDING INTEGRITY

~~1.23 SHIELD BUILDING INTEGRITY shall exist when:~~

~~a. Each door in each access opening is closed except when the access opening is being used for normal transit entry and exit, then at least one door shall be closed,~~

~~b. The shield building filtration system is OPERABLE, and~~

~~c. The sealing mechanism associated with each penetration (e.g., welds, bellows or O-rings) is OPERABLE.~~

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SITE BOUNDARY ← BOP Definition

~~1.29 The SITE BOUNDARY shall be that line beyond which the land is neither owned, nor leased, nor otherwise controlled by the licensee.~~

SOFTWARE

1.30 The digital computer SOFTWARE for the reactor protection system shall be the program codes including their associated data, documentation, and procedures.

SOLIDIFICATION ← BOP Definition

~~1.31 SOLIDIFICATION shall be the conversion of radioactive wastes from liquid systems to a homogeneous (uniformly distributed), monolithic, immobilized solid with definite volume and shape, bounded by a stable surface of distinct outline on all sides (free standing).~~

SOURCE CHECK ← BOP Definition

~~1.32 A SOURCE CHECK shall be the qualitative assessment of channel response when the channel sensor is exposed to a source of increased radioactivity.~~

STAGGERED TEST BASIS

1.33 A STAGGERED TEST BASIS shall consist of:

a. A test schedule for n systems, subsystems, trains, or other designated components obtained by dividing the specified test interval into n equal subintervals, and

b. The testing of one system, subsystem, train, or other designated component at the beginning of each subinterval.

THERMAL POWER

1.34 THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

UNIDENTIFIED LEAKAGE

1.35 UNIDENTIFIED LEAKAGE shall be all leakage which does not constitute either IDENTIFIED LEAKAGE or reactor coolant pump controlled bleed-off flow.

UNRESTRICTED AREA See BOP Definition.

~~1.36 An UNRESTRICTED AREA shall be any area at or beyond the SITE BOUNDARY access to which is not controlled by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials, or any area within the SITE BOUNDARY used for residential quarters or for industrial, commercial, institutional, and/or recreational purposes.~~

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VENTILATION EXHAUST TREATMENT SYSTEM ← See BOP Definition.

~~1.37 A VENTILATION EXHAUST TREATMENT SYSTEM shall be any system designed and installed to reduce gaseous radioiodine or radioactive material in particulate form in effluents by passing ventilation or vent exhaust gases through charcoal adsorbers and/or HEPA filters for the purpose of removing iodines or particulates from the gaseous exhaust stream prior to the release to the environment. Such a system is not considered to have any effect on noble gas effluents. Engineered Safety Feature (ESF) atmospheric cleanup systems are not considered to be VENTILATION EXHAUST TREATMENT SYSTEM components.~~

VENTING ← See BOP Definition.

~~1.38 VENTING shall be the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration, or other operating condition, in such a manner that replacement air or gas is not provided or required during VENTING. Vent, used in system names, does not imply a VENTING process.~~

TABLE 1.1

FREQUENCY NOTATION

NOTATION	FREQUENCY
S	At least once per 12 hours.
D	At least once per 24 hours.
W	At least once per 7 days.
~~4/M~~	~~At least 4 times per month at intervals no greater than 9 days and a minimum of 48 times per year.~~
M	At least once per 31 days.
Q	At least once per 92 days.
SA	At least once per 184 days.
R	At least once per 18 months.
~~P~~	~~Completed prior to each release.~~
S/U	Prior to each reactor startup.
N.A.	Not applicable.

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IMAGE EVALUATION
TEST TARGET (MT-3)

150mm

6"

TABLE 1.2

OPERATIONAL MODES

OPERATIONAL MODE	REACTIVITY CONDITION, K_{eff}	% OF RATED THERMAL POWER*	COLD LEG TEMPERATURE (T_{cold})
1. POWER OPERATION	$\geq$ 0.99	> 5%	$\geq$ 350°F
2. STARTUP	$\geq$ 0.99	$\leq$ 5%	$\geq$ 350°F
3. HOT STANDBY	< 0.99	0	$\geq$ 350°F
4. HOT SHUTDOWN	< 0.99	0	350° > T_{cold} > 210°F
5. COLD SHUTDOWN	< 0.99	0	$\leq$ 210°F
6. REFUELING**	$\leq$ 0.95	0	$\leq$ 135°F

*Excluding decay heat.
**Fuel in the reactor vessel with the vessel head closure bolts less than fully tensioned or with the head removed.

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SECTION 2.0

SAFETY LIMITS

AND

LIMITING SAFETY SYSTEM SETTINGS

CESSAR80-NSSS-STS

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2.1 SAFETY LIMITS

2.1.1 REACTOR CORE

DNBR

2.1.1.1 The calculated DNBR of the reactor core shall be maintained greater than or equal to 1.231.

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the calculated DNBR of the reactor has decreased to less than 1.231, be in HOT STANDBY within 1 hour, and comply with the requirements of Specification 6.7.1.

PEAK LINEAR HEAT RATE

2.1.1.2 The peak linear heat rate (adjusted for fuel rod dynamics) of the fuel shall be maintained less than or equal to 21 kW/ft.

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the peak linear heat rate (adjusted for fuel rod dynamics) of the fuel has exceeded 21 kW/ft, be in HOT STANDBY within 1 hour, and comply with the requirements of Specification 6.7.1.

REACTOR COOLANT SYSTEM PRESSURE

2.1.2 The Reactor Coolant System pressure shall not exceed 2750 psia.

APPLICABILITY: MODES 1, 2, 3, 4, and 5.

ACTION:

MODES 1 and 2:

Whenever the Reactor Coolant System pressure has exceeded 2750 psia, be in HOT STANDBY with the Reactor Coolant System pressure within its limit within 1 hour, and comply with the requirements of Specification 6.7.1.

MODES 3, 4, and 5:

Whenever the Reactor Coolant System pressure has exceeded 2750 psia, reduce the Reactor Coolant System pressure to within its limit within 5 minutes, and comply with the requirements of Specification 6.7.1.

2.2 LIMITING SAFETY SYSTEM SETTINGS

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REACTOR TRIP SETPOINTS

2.2.1 The reactor protective instrumentation setpoints shall be set consistent with the Trip Setpoint values shown in Table 2.2-1.

APPLICABILITY: As shown for each channel in Table 3.3-1.

ACTION:

With a reactor protective instrumentation setpoint less conservative than the value shown in the Allowable Values column of Table 2.2-1, declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3.1 until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

~~CORE PROTECTION CALCULATOR ADDRESSABLE CONSTANTS~~

~~2.2.2 Core Protection Calculator Addressable Constants shall be in accordance with Table 2.2-2.~~

~~APPLICABILITY: As shown for Core Protection Calculators in Table 3.3-1.~~

~~ACTION:~~

~~With a Core Protection Calculator Addressable Constant less conservative than the value shown in the Allowable Value column of Table 2.2-2, declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3.1 until the channel is restored to OPERABLE status.~~

CESSAR 80 - NSSS - STS ~~PALO VERDE - UNIT 1~~

2-3

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TABLE 2.2-1

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
I. TRIP GENERATION		
A. Process		
1. Pressurizer Pressure - High	~~≤ 2383 psia~~ stet	~~≤ 2388 psia~~ stet
2. Pressurizer Pressure - Low (2)	~~≥ 1837 psia (2)~~ *	~~≥ 1822 psia (2)~~ *
3. Steam Generator Level - Low (4)	~~≥ 44.2% (4)~~ *	~~≥ 43.7% (4)~~ *
4. Steam Generator Level - High (9)	~~≤ 91.0% (9)~~ *	~~≤ 91.5% (9)~~ *
5. Steam Generator Pressure - Low (3)	~~≥ 919 psia (3)~~ *	~~≥ 912 psia (3)~~ *
6. Containment Pressure - High	~~≤ 3.0 psig~~ *	~~≤ 3.2 psig~~ *
7. Reactor Coolant Flow - Low (7)		
a. Rate (6)	~~≤ 1.05%/s (6)(7)~~ *	~~≤ 1.10%/s (6)(7)~~ *
b. Floor (6)	~~≥ 52.2% (6)(7)~~ *	~~≥ 47.2% (6)(7)~~ *
c. Band (6)	~~≤ 40.0% (6)(7)~~ *	~~≤ 42.1% (6)(7)~~ *
8. Local Power Density - High	~~≤ 21.0 kW/ft (5)~~ stet	~~≤ 21.0 kW/ft (5)~~ stet
9. DNBR - Low	~~≥ 1.231 (5)~~ stet	~~≥ 1.221 (5)~~ stet
10. Loss of Load (BOP)	*	* (BOP)
B. Excore Neutron Flux		
1. Variable Overpower Trip (10)		
a. Rate (8)	~~≤ 10.6%/min of RATED THERMAL POWER (8)~~ *	~~≤ 11.0%/min of RATED THERMAL POWER (8)~~ *
b. Ceiling (8)	~~≤ 110.0% of RATED THERMAL POWER (8)~~ *	~~≤ 111.0% of RATED THERMAL POWER (8)~~ *
c. Band (8)	~~≤ 9.0% of RATED THERMAL POWER (8)~~ *	~~≤ 10.0% of RATED THERMAL POWER (8)~~ *
11. Linear Power Level (BOP)	*	* (BOP)

* See Applicant's SAR

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CESSAR 80-NSSS-STS

~~PALO VERDE UNIT 1~~

TABLE 2.2-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
2. Logarithmic Power Level - High (1)		
a. Startup and Operating	~~≤ 0.798% of RATED THERMAL POWER~~ *	~~≤ 0.895% of RATED THERMAL POWER~~ *
b. Shutdown	~~≤ 0.798% of RATED THERMAL POWER~~ *	~~≤ 0.895% of RATED THERMAL POWER~~ *
C. Core Protection Calculator System		
1. CEA Calculators	Not Applicable	Not Applicable
2. Core Protection Calculators	Not Applicable	Not Applicable
D. Supplementary Protection System		
Pressurizer Pressure - High	~~≤ 2409 psia~~ *	~~≤ 2414 psia~~ *
II. RPS LOGIC		
A. Matrix Logic	Not Applicable	Not Applicable
B. Initiation Logic	Not Applicable	Not Applicable
III. RPS ACTUATION DEVICES		
A. Reactor Trip Breakers	Not Applicable	Not Applicable
B. Manual Trip	Not Applicable	Not Applicable

* See Applicant's SAR

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TABLE 2.2-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

TABLE NOTATIONS

(1) Trip may be manually bypassed above 10^{-4}% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is less than or equal to 10^{-4}% of RATED THERMAL POWER.

(2) In MODES 3-6, value may be decreased manually, to a minimum of 100 psia, as pressurizer pressure is reduced, provided the margin between the pressurizer pressure and this value is maintained at less than or equal to 400 psi; the setpoint shall be increased automatically as pressurizer pressure is increased until the trip setpoint is reached. Trip may be manually bypassed below 400 psia; bypass shall be automatically removed whenever pressurizer pressure is greater than or equal to 500 psia.

(3) In MODES 3-6, value may be decreased manually as steam generator pressure is reduced, provided the margin between the steam generator pressure and this value is maintained at less than or equal to 200 psi; the setpoint shall be increased automatically as steam generator pressure is increased until the trip setpoint is reached.

(4) % of the distance between steam generator upper and lower level wide range instrument nozzles.

(5) As stored within the Core Protection Calculator (CPC). Calculation of the trip setpoint includes measurement, calculational and processor uncertainties, and dynamic allowances. Trip may be manually bypassed below 1% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is greater than or equal to 1% of RATED THERMAL POWER.

The approved DNBR limit is 1.231 which includes a partial rod bow penalty compensation. If the fuel burnup exceeds that for which an increased rod bow penalty is required, the DNBR limit shall be adjusted. In this case a DNBR trip setpoint of 1.231 is allowed provided that the difference is compensated by an increase in the CPC addressable constant BERR1 as follows:

$$BERR1_{new} = BERR1_{old} \left[1 + \frac{RB - RB_o}{100} \times \frac{d\ (\%\ POL)}{d\ (\%\ DNBR)}\right]$$

where $BERR1_{old}$ is the uncompensated value of BERR1; RB is the fuel rod bow penalty in % DNBR; RB_o is the fuel rod bow penalty in % DNBR already accounted for in the DNBR limit; POL is the power operating limit; and d (% POL)/d (% DNBR) is the absolute value of the most adverse derivative of POL with respect to DNBR.

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TABLE NOTATIONS (Continued)

(6) RATE is the maximum rate of decrease of the trip setpoint. There are no restrictions on the rate at which the setpoint can increase.
FLOOR is the minimum value of the trip setpoint.
BAND is the amount by which the trip setpoint is below the input signal unless limited by Rate or Floor.
Setpoints are % of 100% power flow conditions.

(7) The setpoint may be altered to disable trip function during testing pursuant to Specification 3.10.3.

(8) RATE is the maximum rate of increase of the trip setpoint. There are no restrictions on the rate at which the setpoint can decrease.
CEILING is the maximum value of the trip setpoint.
BAND is the amount by which the trip setpoint is above the input signal unless limited by the rate or the ceiling.

(9) % of the distance between steam generator upper and lower level narrow range instrument nozzles.

(10) % of RATED THERMAL POWER.

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BASES

FOR

SECTION 2.0

SAFETY LIMITS

AND

LIMITING SAFETY SYSTEM SETTINGS

CESSAR80-NSSS-STS

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NOTE

The BASES contained in the succeeding pages summarize the reasons for the specifications of Section 2.0 but in accordance with 10 CFR 50.36 are not a part of these Technical Specifications.

CESSAR80-NSSS-STS

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2.1.1 REACTOR CORE

The restrictions of these safety limits prevent overheating of the fuel cladding and possible cladding perforation which would result in the release of fission products to the reactor coolant. Overheating of the fuel cladding is prevented by (1) restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature, and (2) maintaining the dynamically adjusted peak linear heat rate of the fuel at or less than 21 kW/ft which will not cause fuel centerline melting in any fuel rod.

First, by operating within the nucleate boiling regime of heat transfer, the heat transfer coefficient is large enough so that the maximum clad surface temperature is only slightly greater than the coolant saturation temperature. The upper boundary of the nucleate boiling regime is termed "departure from nucleate boiling" (DNB). At this point, there is a sharp reduction of the heat transfer coefficient, which would result in higher cladding temperatures and the possibility of cladding failure.

Correlations predict DNB and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB ratio (DNBR), defined as the ratio of the predicted DNB heat flux at a particular core location to the actual heat flux at that location, is indicative of the margin to DNB. The minimum value of DNBR during normal operation and design basis anticipated operational occurrences is limited to 1.231 based upon a statistical combination of CE-1 CHF correlation and engineering factor uncertainties and is established as a Safety Limit. ~~The DNBR limit of 1.231 includes a rod bow compensation of 0.8% on DNBR. For fuel burnups which exceed that for which an increased rod bow penalty is required, the DNBR limit shall be adjusted. In this case the DNBR trip setpoint of 1.231 is allowed if the required DNBR increase is compensated by an increase of the addressable constant BERR1.~~

Second, operation with a peak linear heat rate below that which would cause fuel centerline melting maintains fuel rod and cladding integrity. Above this peak linear heat rate level (i.e., with some melting in the center), fuel rod integrity would be maintained only if the design and operating conditions are appropriate throughout the life of the fuel rods. Volume changes which accompany the solid to liquid phase change are significant and require accommodation. Another consideration involves the redistribution of the fuel which depends on the extent of the melting and the physical state of the fuel rod at the time of melting. Because of the above factors, the steady state value of the peak linear heat rate which would not cause fuel centerline melting is established as a Safety Limit. To account for fuel rod dynamics (lags), the directly indicated linear heat rate is dynamically adjusted by the CPC program.

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BASES

Limiting Safety System Settings for the Low DNBR, High Local Power Density, High Logarithmic Power Level, Low Pressurizer Pressure and High Linear Power Level trips, and Limiting Conditions for Operation on DNBR and kW/ft margin are specified such that there is a high degree of confidence that the specified acceptable fuel design limits are not exceeded during normal operation and design basis anticipated operational occurrences.

2.1.2 REACTOR COOLANT SYSTEM PRESSURE

The restriction of this Safety Limit protects the integrity of the Reactor Coolant System from overpressurization and thereby prevents the release of radionuclides contained in the reactor coolant from reaching the containment atmosphere.

~~The Reactor Coolant System components are designed to Section III, 1974 Edition, Summer 1975 Addendum, of the ASME Code for Nuclear Power Plant Components which permits a maximum transient pressure of 110% (2750 psia) of design pressure. The Safety Limit of 2750 psia is therefore consistent with the design criteria and associated code requirements.~~ Insert A

The entire Reactor Coolant System is hydrotested at 3125 psia to demonstrate integrity prior to initial operation.

2.2.1 REACTOR TRIP SETPOINTS

The Reactor Trip Setpoints specified in Table 2.2-1 are the values at which the Reactor Trips are set for each functional unit. The Trip Setpoints have been selected to ensure that the reactor core and Reactor Coolant System are prevented from exceeding their Safety Limits during normal operation and design basis anticipated operational occurrences and to assist the Engineered Safety Features Actuation System in mitigating the consequences of accidents. Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is equal to or less than the drift allowance assumed for each trip in the safety analyses.

The DNBR - Low and Local Power Density - High are digitally generated trip setpoints based on Safety Limits of 1.231 and 21 kW/ft, respectively. Since these trips are digitally generated by the Core Protection Calculators, the trip values are not subject to drifts common to trips generated by analog type equipment. The Allowable Values for these trips are therefore the same as the Trip Setpoints.

To maintain the margins of safety assumed in the safety analyses, the calculations of the trip variables for the DNBR - Low and Local Power Density - High trips include the measurement, calculational and processor uncertainties and dynamic allowances as defined in CESSAR System 80 applicable system descriptions and safety analyses.

Insert

~~2.1.2 REACTOR COOLANT SYSTEM PRESSURE~~

~~The restriction of this safety limit protects the integrity of the Reactor Coolant System from overpressurization and thereby prevents the release of radionuclides contained in the reactor coolant from reaching the containment atmosphere.~~

(A) The reactor pressure vessel, piping, and pressurizer are designed to Section III of the ASME Code for Nuclear Power Plant Components which permits a maximum transient pressure of 110% (2750 psia) of the design pressure. The Reactor Coolant System valves and fittings, are designed to either Section III of the ASME Code or ANSI B 31.7, Class I, which permits a maximum transient pressure of 110% (2750 psia) of component design pressure. See Applicant's FSAR for specific Code, Standard Editions, and Addenda. The safety limit of 2750 psia is therefore consistent with the design criteria and associated code requirements.

~~The entire Reactor Coolant System is hydrotested at 3125 psia to demonstrate integrity prior to initial operation.~~

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Amendment Number 9
February 27, 1984

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REACTOR TRIP SETPOINTS (Continued)

~~The methodology for the calculation of the PVNGS trip setpoint values, plant protection system, is discussed in the CE Document No. CEN-286(V) dated July 31, 1984.~~

Manual Reactor Trip

The Manual reactor trip is a redundant channel to the automatic protective instrumentation channels and provides manual reactor trip capability.

Variable Overpower Trip

A reactor trip on Variable Overpower is provided to protect the reactor core during rapid positive reactivity addition excursions. This trip function will trip the reactor when the indicated neutron flux power exceeds either a rate limited setpoint at a great enough rate or reaches a preset ceiling. The flux signal used is the average of three linear subchannel flux signals originating in each nuclear instrument safety channel. These trip setpoints are provided in Table 2.2-1.

Logarithmic Power Level - High

The Logarithmic Power Level - High trip is provided to protect the integrity of fuel cladding and the Reactor Coolant System pressure boundary in the event of an unplanned criticality from a shutdown condition. A reactor trip is initiated by the Logarithmic Power Level - High trip unless this trip is manually bypassed by the operator. The operator may manually bypass this trip when the THERMAL POWER level is above 10^{-4}% of RATED THERMAL POWER; this bypass is automatically removed when the THERMAL POWER level decreases to 10^{-4}% of RATED THERMAL POWER.

Pressurizer Pressure - High

The Pressurizer Pressure - High trip, in conjunction with the pressurizer safety valves and main steam safety valves, provides Reactor Coolant System protection against overpressurization in the event of loss of load without reactor trip. This trip's setpoint is below the nominal lift setting of the pressurizer safety valves and its operation minimizes the undesirable operation of the pressurizer safety valves.

Pressurizer Pressure - Low

The Pressurizer Pressure - Low trip is provided to trip the reactor and to assist the Engineered Safety Features System in the event of a decrease in Reactor Coolant System inventory and in the event of an increase in heat

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Pressurizer Pressure - Low (Continued)

removal by the secondary system. During normal operation, this trip's setpoint may be manually decreased, to a minimum value of 100 psia, as pressurizer pressure is reduced during plant shutdowns, provided the margin between the pressurizer pressure and this trip's setpoint is maintained at less than or equal to 400 psi; this setpoint increases automatically as pressurizer pressure increases until the trip setpoint is reached. The operator may manually bypass this trip when pressurizer pressure is below 400 psia. This bypass is automatically removed when the pressurizer pressure increases to 500 psia.

Containment Pressure - High

The Containment Pressure - High trip provides assurance that a reactor trip is initiated in the event of containment building pressurization due to a pipe break inside the containment building. The setpoint for this trip is identical to the safety injection setpoint.

Steam Generator Pressure - Low

The Steam Generator Pressure - Low trip provides protection in the event of an increase in heat removal by the secondary system and subsequent cooldown of the reactor coolant. The setpoint is sufficiently below the full load operating point so as not to interfere with normal operation, but still high enough to provide the required protection in the event of excessively high steam flow. This trip's setpoint may be manually decreased as steam generator pressure is reduced during plant shutdowns, provided the margin between the steam generator pressure and this trip's setpoint is maintained at less than or equal to 200 psi; this setpoint increases automatically as steam generator pressure increases until the normal pressure trip setpoint is reached.

Steam Generator Level - Low

The Steam Generator Level - Low trip provides protection against a loss of feedwater flow incident and assures that the design pressure of the Reactor Coolant System will not be exceeded due to a decrease in heat removal by the secondary system. This specified setpoint provides allowance that there will be sufficient water inventory in the steam generator at the time of the trip to provide a margin of at least 10 minutes before auxiliary feedwater is required to prevent degraded core cooling.

Local Power Density - High

The Local Power Density - High trip is provided to prevent the linear heat rate (kW/ft) in the limiting fuel rod in the core from exceeding the fuel design limit in the event of any ~~design bases~~ anticipated operational occurrence. The local power density is calculated in the reactor protective system utilizing the following information:

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Local Power Density - High (Continued)

a. Nuclear flux power and axial power distribution from the excore flux monitoring system;

b. Radial peaking factors from the position measurement for the CEAs;

c. Delta T power from reactor coolant temperatures and coolant flow measurements.

The local power density (LPD), the trip variable, calculated by the CPC incorporates uncertainties and dynamic compensation routines. These uncertainties and dynamic compensation routines ensure that a reactor trip occurs when the actual core peak LPD is sufficiently less than the fuel design limit such that the increase in actual core peak LPD after the trip will not result in a violation of the Peak Linear Heat Rate Safety Limit. CPC uncertainties related to peak LPD are the same types used for DNBR calculation. Dynamic compensation for peak LPD is provided for the effects of core fuel centerline temperature delays (relative to changes in power density), sensor time delays, and protection system equipment time delays.

DNBR - Low

The DNBR - Low trip is provided to prevent the DNBR in the limiting coolant channel in the core from exceeding the fuel design limit in the event of ~~design bases~~ anticipated operational occurrences. The DNBR - Low trip incorporates a low pressurizer pressure floor of [illegible] psia. (*) At this pressure a DNBR - Low trip will automatically occur. The DNBR is calculated in the CPC utilizing the following information:

a. Nuclear flux power and axial power distribution from the excore neutron flux monitoring system;

b. Reactor Coolant System pressure from pressurizer pressure measurement;

c. Differential temperature (Delta T) power from reactor coolant temperature and coolant flow measurements;

d. Radial peaking factors from the position measurement for the CEAs;

e. Reactor coolant mass flow rate from reactor coolant pump speed;

f. Core inlet temperature from reactor coolant cold leg temperature measurements.

* See Applicant's SAR

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DNBR - Low (Continued)

The DNBR, the trip variable, calculated by the CPC incorporates various uncertainties and dynamic compensation routines to assure a trip is initiated prior to violation of fuel design limits. These uncertainties and dynamic compensation routines ensure that a reactor trip occurs when the calculated core DNBR is sufficiently greater than 1.231 such that the decrease in calculated core DNBR after the trip will not result in a violation of the DNBR Safety Limit. CPC uncertainties related to DNBR cover CPC input measurement uncertainties, algorithm modelling uncertainties, and computer equipment processing uncertainties. Dynamic compensation is provided in the CPC calculations for the effects of coolant transport delays, core heat flux delays (relative to changes in core power), sensor time delays, and protection system equipment time delays.

The DNBR algorithm used in the CPC is valid only within the limits indicated below and operation outside of these limits will result in a CPC initiated trip.

	Parameter	Limiting Value*
a.	RCS Cold Leg Temperature-Low	≥ [redacted]°F
b.	RCS Cold Leg Temperature-High	≤ [redacted]°F
c.	Axial Shape Index-Positive	Not more positive than [redacted]
d.	Axial Shape Index-Negative	Not more negative than [redacted]
e.	Pressurizer Pressure-Low	≥ [redacted] psia
f.	Pressurizer Pressure-High	≤ [redacted] psia
g.	Integrated Radial Peaking Factor-Low	≥ [redacted]
h.	Integrated Radial Peaking Factor-High	≤ [redacted]
i.	Quality Margin-Low	≥ [redacted]

Steam Generator Level - High

The Steam Generator Level - High trip ~~is provided to protect the turbine from excessive moisture carry over. Since the turbine is automatically tripped when the reactor is tripped, this trip provides a reliable means for providing protection to the turbine from excessive moisture carryover. This trip's setpoint does not correspond to a safety limit, and provides protection in the event of excess feedwater flow.~~ provides protection in the event of excess feedwater flow. The setpoint is identical to the main steam isolation setpoint. ~~Its functional capability at the specified trip setting enhances the overall reliability of the reactor protection system.~~

* See Applicant's SAR

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two-pump opposite loop flow event.

Reactor Coolant Flow - Low

The Reactor Coolant Flow - Low trip provides protection against a reactor coolant pump sheared shaft event and a ~~four pump flow~~ coastdown ~~during a steam line break with loss of offsite power~~. A trip is initiated when the pressure differential across the primary side of either steam generator decreases below a variable setpoint. This variable setpoint stays a set amount below the pressure differential unless limited by a set maximum decrease rate or a set minimum value. The specified setpoint ensures that a reactor trip occurs ~~to prevent violation of Peak Linear Heat Rate or DNBR Safety Limits under the stated conditions~~ consistent with the safety analysis.

Pressurizer Pressure - High (SPS)

The Supplementary Protection System (SPS) augments reactor protection against overpressurization by utilizing a separate and diverse trip logic from the Reactor Protection System for initiation of reactor trip. The SPS will initiate a reactor trip when pressurizer pressure exceeds a predetermined value.

Loss of Load and Linear Power Level (BOP)

R See Applicant's SAR.

SECTIONS 3.0 AND 4.0

LIMITING CONDITIONS FOR OPERATION

AND

SURVEILLANCE REQUIREMENTS

CESSAR80-NSSS-STS

3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

3/4.0 APPLICABILITY

LIMITING CONDITION FOR OPERATION

3.0.1 Compliance with the Limiting Conditions for Operation contained in the succeeding specifications is required during the OPERATIONAL MODES or other conditions specified therein; except that upon failure to meet the Limiting Conditions for Operation, the associated ACTION requirements shall be met.

3.0.2 Noncompliance with a specification shall exist when the requirements of the Limiting Condition for Operation and/or associated ACTION requirements are not met within the specified time intervals. If the Limiting Condition for Operation is restored prior to expiration of the specified time intervals, completion of the ACTION requirements is not required.

3.0.3 When a Limiting Condition for Operation is not met, except as provided in the associated ACTION requirements, within 1 hour, action shall be initiated to place the unit in a MODE in which the specification does not apply by placing it, as applicable, in:

1. At least HOT STANDBY within the next 6 hours,
2. At least HOT SHUTDOWN within the following 6 hours, and
3. At least COLD SHUTDOWN within the subsequent 24 hours.

Where corrective measures are completed that permit operation under the ACTION requirements, the ACTION may be taken in accordance with the specified time limits as measured from the time of failure to meet the Limiting Condition for Operation. Exceptions to these requirements are stated in the individual specifications.

This specification is not applicable in MODE 5 or 6.

3.0.4 Entry into an OPERATIONAL MODE or other specified condition shall not be made unless the conditions of the Limiting Condition for Operation are met without reliance on provisions contained in the ACTION requirements. This provision shall not prevent passage through or to OPERATIONAL MODES as required to comply with ACTION statements. Exceptions to these requirements are stated in the individual specifications.

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SURVEILLANCE REQUIREMENTS

4.0.1 Surveillance Requirements shall be applicable 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 three 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 time interval shall constitute a failure to meet the OPERABILITY requirements for a Limiting Condition for Operation. Exceptions to these requirements are stated in the individual specifications. Surveillance Requirements do not have to be performed on inoperable equipment.

4.0.4 Entry into an OPERATIONAL MODE or other specified condition shall not be made unless the Surveillance Requirement(s) associated with the Limiting Condition for Operation have been performed within the stated surveillance interval or as otherwise specified.

4.0.5 Surveillance Requirements for inservice inspection and testing of ASME Code Class 1, 2, and 3 components shall be applicable as follows:

a. Inservice inspection of ASME Code Class 1, 2, and 3 components and inservice testing ASME Code Class 1, 2, and 3 pumps and valves shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50, Section 50.55a(g), except where specific written relief has been granted by the Commission pursuant to 10 CFR 50, Section 50.55a(g)(6)(i).

b. Surveillance intervals specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda for the inservice inspection and testing activities required by the ASME Boiler and Pressure Vessel Code and applicable Addenda shall be applicable as follows in these Technical Specifications:

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SURVEILLANCE REQUIREMENTS (Continued)

4.0.5 (Continued)

ASME Boiler and Pressure Vessel Code and applicable Addenda terminology for inservice inspection and testing activities	Required frequencies for performing inservice inspection and testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Yearly or annually	At least once per 366 days

c. The provisions of Specification 4.0.2 are applicable to the above required frequencies for performing inservice inspection and testing activities.

d. Performance of the above inservice inspection and testing activities shall be in addition to other specified Surveillance Requirements.

e. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any Technical Specification.

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - T_{cold} GREATER THAN 210°F

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be greater than or equal to 6.0% delta k/k.

APPLICABILITY: MODES 1, 2*, 3, and 4.

ACTION:

With the SHUTDOWN MARGIN less than 6.0% delta k/k, immediately initiate and continue boration at greater than or equal to 26 gpm to reactor coolant system of a solution containing greater than or equal to 4000 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be greater than or equal to 6.0% delta k/k:

a. Within 1 hour after detection of an inoperable CEA(s) and at least once per 12 hours thereafter while the CEA(s) is inoperable. If the inoperable CEA is immovable as a result of excessive friction or mechanical interference or known to be untrippable, the above required SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable CEA(s).

b. When in MODE 1 or MODE 2 with K_{eff} greater than or equal to 1.0, at least once per 12 hours by verifying that CEA group withdrawal is within the Transient Insertion Limits of Specification 3.1.3.6.

c. When in MODE 2 with K_{eff} less than 1.0, within 4 hours prior to achieving reactor criticality by verifying that the predicted critical CEA position is within the limits of Specification 3.1.3.6.

* See Special Test Exception 3.10.1.

SURVEILLANCE REQUIREMENTS (Continued)

d. Prior to initial operation above 5% RATED THERMAL POWER after each fuel loading, by consideration of the factors of e. below, with the CEA groups at the Transient Insertion Limits of Specification 3.1.3.6.

e. When in MODE 3 or 4, at least once per 24 hours by consideration of at least the following factors:

1. Reactor Coolant System boron concentration,
2. CEA position,
3. Reactor Coolant System average temperature,
4. Fuel burnup based on gross thermal energy generation,
5. Xenon concentration, and
6. Samarium concentration.

4.1.1.1.2 The overall core reactivity balance shall be compared to predicted values to demonstrate agreement within $\pm$ 1.0% delta k/k at least once per 31 Effective Full Power Days (EFPD). This comparison shall consider at least those factors stated in Specification 4.1.1.1.1e., above. The predicted reactivity values shall be adjusted (normalized) to correspond to the actual core conditions prior to exceeding a fuel burnup of 60 EFPD after each fuel loading.

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SHUTDOWN MARGIN - T_{cold} LESS THAN OR EQUAL TO 210°F

LIMITING CONDITION FOR OPERATION

3.1.1.2 The SHUTDOWN MARGIN shall be greater than or equal to 4.0% delta k/k.

APPLICABILITY: MODE 5.

ACTION:

With the SHUTDOWN MARGIN less than 4.0% delta k/k, immediately initiate and continue boration at greater than or equal to 26 gpm to the reactor coolant system of a solution containing greater than or equal to 4000 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.2 The SHUTDOWN MARGIN shall be determined to be greater than or equal to 4.0% delta k/k:

a. Within 1 hour after detection of an inoperable CEA(s) and at least once per 12 hours thereafter while the CEA(s) is inoperable. If the inoperable CEA is immovable as a result of excessive friction or mechanical interference or known to be untrippable, the above required SHUTDOWN MARGIN shall be increased by an amount at least equal to the withdrawn worth of the immovable or untrippable CEA(s).

b. At least once per 24 hours by consideration of the following factors:

1. Reactor Coolant System boron concentration,
2. CEA position,
3. Reactor Coolant System average temperature,
4. Fuel burnup based on gross thermal energy generation,
5. Xenon concentration, and
6. Samarium concentration.

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MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.3 The moderator temperature coefficient (MTC) shall be within the area of Acceptable Operation shown on Figure 3.1-1.

APPLICABILITY: MODES 1 and 2*#

ACTION:

With the moderator temperature coefficient outside the area of Acceptable Operation shown on Figure 3.1-1, be in at least HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.1.3.1 The MTC shall be determined to be within its limits by confirmatory measurements. MTC measured values shall be extrapolated and/or compensated to permit direct comparison with the above limits.

4.1.1.3.2 The MTC shall be determined at the following frequencies and THERMAL POWER conditions during each fuel cycle:

a. Prior to initial operation above 5% of RATED THERMAL POWER, after each fuel loading.

b. At any THERMAL POWER, within 7 EFPD after reaching a core average exposure of 40 EFPD burnup into the current cycle.

c. At any THERMAL POWER, within 7 EFPD after reaching a core average exposure equivalent to two-thirds of the expected current cycle end-of-cycle core average burnup.

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*With Keff greater than or equal to 1.0.

#See Special Test Exception 3.10.2.

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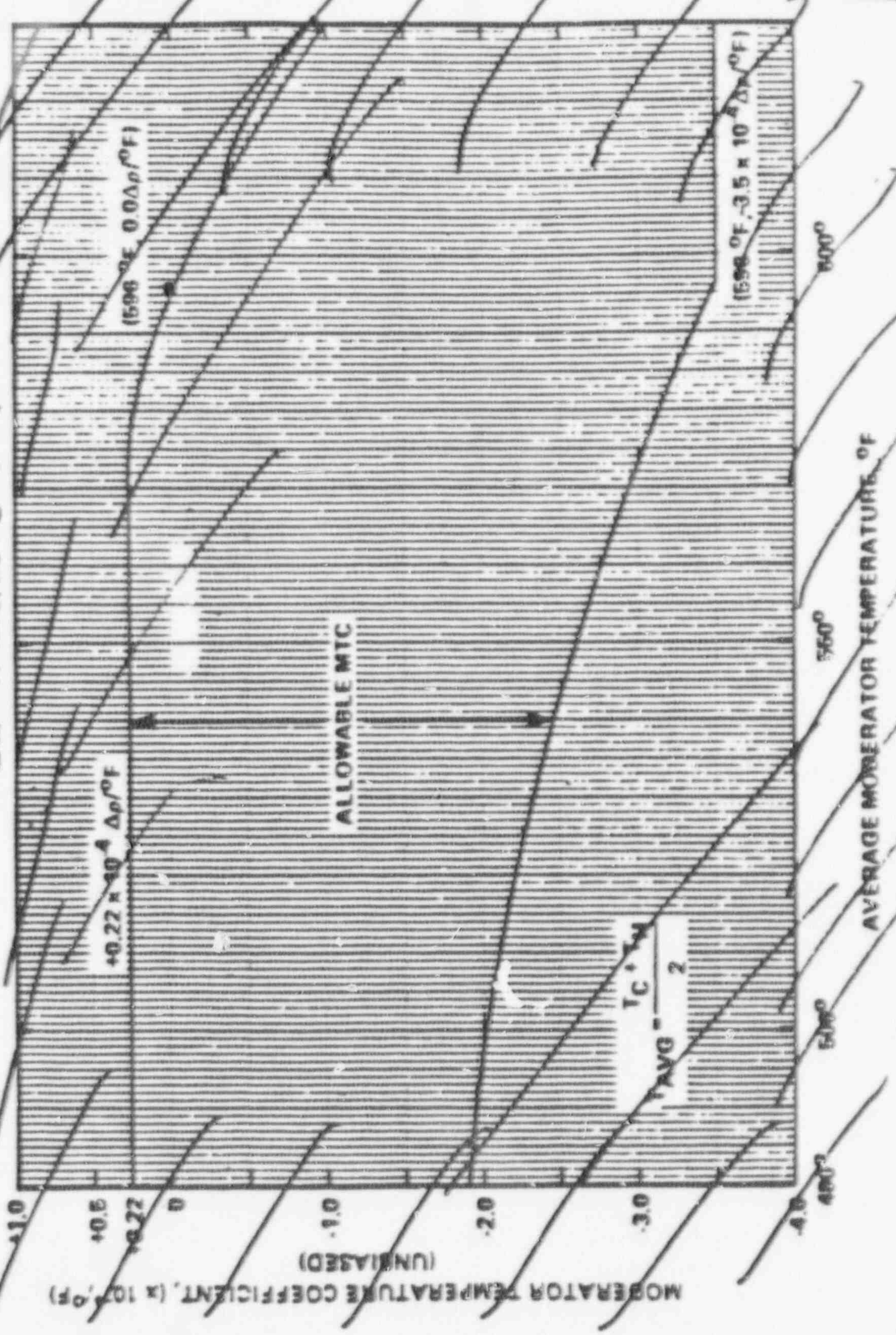

FIGURE 3.1-1

ALLOWABLE MTC MODES 1 AND 2

~~PALO VERDE UNIT 1 CYCLE 1~~

SEE APPLICANTS SAR

MINIMUM TEMPERATURE FOR CRITICALITY

LIMITING CONDITION FOR OPERATION

3.1.1.4 The Reactor Coolant System lowest operating loop temperature (T_{cold}) shall be greater than or equal to 552°F.

APPLICABILITY: MODES 1 and 2#*.

ACTION:

With a Reactor Coolant System operating loop temperature (T_{cold}) less than 552°F, restore T_{cold} to within its limit within 15 minutes or be in HOT STANDBY within the next 15 minutes.

SURVEILLANCE REQUIREMENTS

4.1.1.4 The Reactor Coolant System temperature (T_{cold}) shall be determined to be greater than or equal to 552°F:

a. Within 15 minutes prior to achieving reactor criticality, and

b. At least once per 30 minutes when the reactor is critical and the Reactor Coolant System T_{cold} is less than 557°F.

#With K_{eff} greater than or equal to 1.0.

*See Special Test Exception 3.10.5.

3/4.1.2 BORATION SYSTEMS

FLOW PATHS - SHUTDOWN

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LIMITING CONDITION FOR OPERATION

3.1.2.1 As a minimum, one of the following boron injection flow paths shall be OPERABLE:

a. If only the spent fuel pool in Specification 3.1.2.5a. is OPERABLE, a flow path from the spent fuel pool via a gravity feed connection and a charging pump to the Reactor Coolant System.

b. If only the refueling water tank in Specification 3.1.2.5b. is OPERABLE, a flow path from the refueling water tank via either a charging pump, a high pressure safety injection pump, or a low pressure safety injection pump to the Reactor Coolant System.

APPLICABILITY: MODES 5 and 6.

ACTION:

With none of the above flow paths OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

SURVEILLANCE REQUIREMENTS

4.1.2.1 At least one of the above required flow paths shall be demonstrated OPERABLE at least once per 31 days by verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

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FLOW PATHS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.2 At least two of the following three boron injection flow paths shall be OPERABLE:

a. A gravity feed flow path from either the refueling water tank or the spent fuel pool through CH-536 (RWT Gravity Feed Isolation Valve) and a charging pump to the Reactor Coolant System,

b. A gravity feed flow path from the refueling water tank through CH-327 (RWT Gravity Feed/Safety Injection System Isolation Valve) and a charging pump to the Reactor Coolant System,

c. A flow path from either the refueling water tank or the spent fuel pool through CH-164 (Boric Acid Filter Bypass Valve), utilizing gravity feed and a charging pump to the Reactor Coolant System.

APPLICABILITY: MODES 1, 2, 3, and 4.

CH-161 (Boric Acid Filter Isolation Valve) or

ACTION:

With only one of the above required boron injection flow paths to the Reactor Coolant System OPERABLE, restore at least two boron injection flow paths to the Reactor Coolant System to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least 6% delta k/k at 210°F within the next 6 hours; restore at least two flow paths to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.2 At least two of the above required flow paths shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

b. At least once per 18 months when the Reactor Coolant System is at normal operating pressure by verifying that the flow path required by Specification 3.1.2.2 delivers at least 26 gpm for 1 charging pump and 68 gpm for two charging pumps to the Reactor Coolant System.

~~4.1.2.3 The provisions of Specification 4.0.4 are not applicable for entry into Mode 3 or Mode 4 to perform the surveillance testing of Specification 4.1.2.2.b provided the testing is performed within 24 hours after achieving normal operating pressure in the reactor coolant system.~~

CHARGING PUMPS - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.3 At least one charging pump* or one high pressure safety injection pump or one low pressure safety injection pump in the boron injection flow path required OPERABLE pursuant to Specification 3.1.2.1 shall be OPERABLE and capable of being powered from an OPERABLE emergency power source.

APPLICABILITY: MODES 5 and 6.

ACTION:

With no charging pump or high pressure safety injection pump or low pressure safety injection pump OPERABLE or capable of being powered from an OPERABLE emergency power source, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

SURVEILLANCE REQUIREMENTS

4.1.2.3 No additional Surveillance Requirements other than those required by Specification 4.0.5.

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*Whenever the reactor coolant level is below the bottom of the pressurizer in MODE 5, one and only one charging pump shall be OPERABLE, by verifying at least once per every 7 days that power is removed from the remaining charging pumps.

CHARGING PUMPS - OPERATING

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LIMITING CONDITION FOR OPERATION

3.1.2.4 At least two charging pumps shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With only one charging pump OPERABLE, restore at least two charging pumps to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least 6% delta k/k at 210°F within the next 6 hours; restore at least two charging pumps to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.4 No additional Surveillance Requirements other than those required by Specification 4.0.5.

BORATED WATER SOURCES - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.5 As a minimum, one of the following borated water sources shall be OPERABLE:

a. The spent fuel pool with:

1. A minimum borated water volume ~~of 33,600 gallons~~ as specified in Figure 3.1-2, and
2. A boron concentration of between 4000 ppm and 4400 ppm boron, and
3. A solution temperature between 60°F and 180°F.

b. The refueling water tank with:

1. A minimum contained borated water volume ~~of 33,600 gallons~~ as specified in Figure 3.1-2 and
2. A boron concentration of between 4000 ppm and 4400 ppm boron, and
3. A solution temperature between 60°F and 120°F.

APPLICABILITY: MODES 5~~#~~ and 6~~#~~.

ACTION:

With no borated water sources OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes until at least one borated water source is restored to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.1.2.5 The above required borated water sources shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1. Verifying the boron concentration of the water, and
2. Verifying the contained borated water volume of the refueling water tank or the spent fuel pool.

b. At least once per 24 hours by verifying the refueling water tank temperature when it is the source of borated water and the outside air temperature is outside the 60°F to 120°F range.

c. At least once per 24 hours by verifying the spent fuel pool temperature when it is the source of borated water and irradiated fuel is present in the pool.

~~#See Special Test Exception 3.10.7.~~

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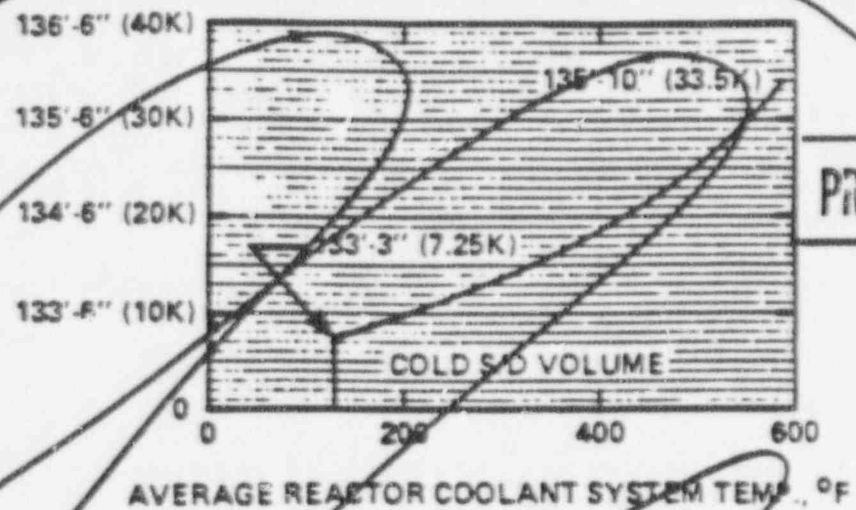

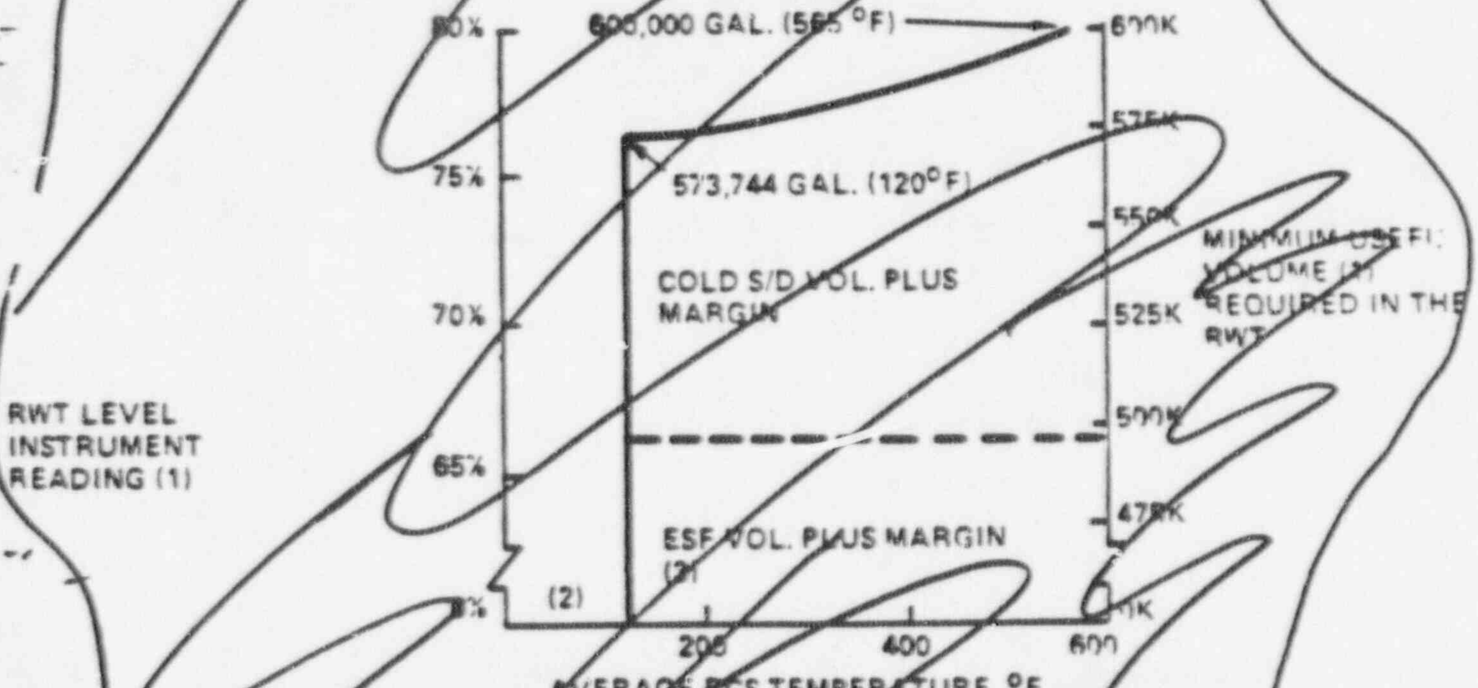

(1) THE TANK LEVEL AND VOLUME SHOWN ARE THE USEFUL LEVEL AND VOLUME ABOVE THAT IN THE TANK WHICH IS REQUIRED FOR VORTEX CONSIDERATIONS

(2) DURING MODE 5 AND 6 ONE OF THESE BORATED SOURCES SHALL CONTAIN A MINIMUM OF 33,500 GALLONS

(3) THIS VOLUME IS NOT REQUIRED DURING MODE 6

FIGURE 3.1-2
MINIMUM BORATED WATER VOLUMES

SEE APPLICANTS SAR

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BORATED WATER SOURCES - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.6 Each of the following borated water sources shall be OPERABLE:

a. The spent fuel pool with:

1. A minimum borated water volume as specified in Figure 3.1-2, and
2. A boron concentration of between 4000 ppm and 440 ppm boron, and
3. A solution temperature between 60°F and 180°F.

b. The refueling water tank with:

1. A minimum contained borated water volume as specified in Figure 3.1-2, and
2. A boron concentration of between 4000 and 4400 ppm of boron, and
3. A solution temperature between 60°F and 120°F.

APPLICABILITY: MODES 1, 2,* 3,* and 4*.

ACTION:

a. With the above required spent fuel pool inoperable, restore the pool to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and borated to a SHUTDOWN MARGIN equivalent to at least 6% delta k/k at 210°F, restore the above required spent fuel poo to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

b. With the refueling water tank inoperable, restore the tank to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.6 Each of the above required borated water sources shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1. Verifying the boron concentration in the water, and
2. Verifying the contained borated water volume of the water source.

b. At least once per 24 hours by verifying the refueling water tank temperature when the outside air temperature is outside the 60°F to 120°F range.

c. At least once per 24 hours by verifying the spent fuel pool temperature when irradiated fuel is present in the pool.

~~*See Special Test Exception 3.10.7~~

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LIMITING CONDITION FOR OPERATION

3.1.2.7 Both startup channel high neutron flux alarms shall be OPERABLE.

APPLICABILITY: MODES 3*, 4, 5, and 6.

ACTION:

a. With one startup channel high neutron flux alarm inoperable:

1. Determine the RCS boron concentration when entering MODE 3, 4, 5, or 6 or at the time the alarm is determined to be inoperable. From that time, the RCS boron concentration shall be determined at the applicable monitoring frequency in Tables 3.1-1 through 3.1-5 by either boronometer or RCS sampling.**

b. With both startup channel high neutron flux alarms inoperable:

1. Determine the RCS boron concentration by either boronmeter and RCS sampling** or by independent collection and analysis of two RCS samples when entering Mode 3, 4, or 5 or at the time both alarms are determined to be inoperable. From that time, the RCS boron concentration shall be determined at the applicable monitoring frequency in Tables 3.1-1 through 3.1-5, as applicable, by either boronmeter and RCS sampling** or by collection and analysis of two independent RCS samples. If redundant determination of RCS boron concentration cannot be accomplished immediately, suspend all operations involving CORE ALTERATIONS or positive reactivity changes until the method for determining and confirming RCS boron concentration is restored.

2. When in MODE 5 with the RCS level below the centerline of the hotleg or MODE 6, suspend all operations involving CORE ALTERATIONS or positive reactivity changes until at least one startup channel high neutron flux alarm is restored to OPERABLE status.

c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.1.2.7 Each startup channel high neutron flux alarm shall be demonstrated OPERABLE by performance of:

---

*Within 1 hour after the neutron flux is within the startup range following a reactor shutdown.

**With one or more reactor coolant pumps (RCP) operating the sample should be obtained from the hot leg. With no RCP operating, the sample should be obtained from the discharge line of the low pressure safety injection (LPSI) pump operating in the shutdown cooling mode.

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SURVEILLANCE REQUIREMENTS (Continued)

a. A CHANNEL CHECK:

1. At least once per 12 hours.

2. When initially setting setpoints at the following times:

a) One hour after a reactor trip.
b) After a controlled reactor shutdown: Within 1 hour after the neutron flux is within the startup range in MODE 3.

b. A CHANNEL FUNCTIONAL TEST every 31 days of cumulative operation during shutdown.

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TABLE 3.1-1

~~REQUIRED~~ MONITORING FREQUENCIES FOR BACKUP BORON DILUTION DETECTION ~~AS A FUNCTION OF OPERATING CHARGING PUMPS AND PLANT OPERATIONAL MODES~~ FOR $K_{eff} > 0.98$

FOR SYSTEM 80 EXTENDED FUEL CYCLE

OPERATIONAL MODE	Number of Operating Charging Pumps 0	1	2	3
3	12 hours	1 hour	Operation not allowed	
4	12 hours	1 hour	Operation not allowed	
5 RCS filled	8 hours	1 hour	Operation not allowed	
5 RCS partially drained	Operation not allowed			
6	24 hours	8 hours	4 hours	2 hours

TABLE 3.1-2

~~REQUIRED~~ MONITORING FREQUENCIES FOR BACKUP BORON DILUTION DETECTION ~~AS A FUNCTION OF OPERATING CHARGING PUMPS AND PLANT OPERATIONAL MODES~~ FOR 0.98 $\geq K_{eff} > 0.97$

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OPERATIONAL MODE	Number of Operating Charging Pumps 0	1	2	3
3	12 hours	2.5 hours	1 hour	0.5 hours
4	12 hours	2.5 hours	1 hour	0.5 hours
5 RCS filled	8 hours	2.5 hours	1 hour	0.5 hours
5 RCS partially drained*	8 hours	0.5 hours	Operation not allowed	
6	24 hours	8 hours	4 hours	2 hours

* The Technical Specification 3.1.2.3 will allow operation of only ONE charging pump during this MODE and plant condition.

FOR SYSTEM 80 EXTENDED FUEL CYCLE [redacted]

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TABLE 3.1-3

[redacted]D MONITORING FREQUENCIES FOR BACKUP BORON DILUTION DETECTION ~~AS A FUNCTION OF OPERATING CHARGING PUMPS~~ ~~AND PLANT OPERATIONAL MODES~~ FOR $0.97 \geq K_{eff} > 0.96$

FOR SYSTEM 80 EXTENDED FUEL [redacted] CYCLE

OPERATIONAL MODE	Number of Operating Charging Pumps 0	1	2	3
3	12 hours	3.5 hours	1.5 hours	1 hour
4	12 hours	3.5 hours	1.5 hours	1 hour
5 RCS filled	8 hours	3.5 hours	1.5 hours	1 hour
5 RCS partially drained*	8 hours	1 hour	Operation not allowed	
6	24 hours	8 hours	4 hours	2 hours

* The [redacted] Technical Specification 3.1.2.3 will allow operation of only ONE charging pump during this MODE and plant condition.

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TABLE 3.1-4

[redacted] MONITORING FREQUENCIES FOR BACKUP BORON DILUTION DETECTION ~~AS A FUNCTION OF OPERATING CHARGING PUMPS, AND PLANT OPERATIONAL MODES~~ FOR $0.96 \geq K_{eff} > 0.95$

OPERATIONAL MODE	Number of Operating Charging Pumps 0	1	2	3
3	12 hours	5 hours	2 hours	1 hour
4	12 hours	5 hours	2 hours	1 hour
5 RCS filled	8 hours	5 hours	2 hours	1 hour
5 RCS partially drained*	8 hours	1.5 hours	Operation not allowed	
6	24 hours	8 hours	4 hours	2 hours

* The Technical Specification 3.1.2.3 will allow operation of only ONE charging pump this MODE and plant condition.

FOR SYSTEM 80 EXTENDED [redacted] FUEL CYCLE

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TABLE 3.1-5

~~REQUIRED~~ MONITORING FREQUENCIES FOR BACKUP BORON DILUTION DETECTION ~~AS A FUNCTION OF OPERATING CHARGING PUMPS AND PLANT OPERATIONAL MODES~~ FOR $K_{eff} \leq 0.95$

OPERATIONAL MODE	Number of Operating Charging Pumps 0	1	2	3
3	12 hours	6 hours	3 hours	1.5 hours
4	12 hours	6 hours	3 hours	1.5 hours
5 RCS filled	8 hours	6 hours	3 hours	1.5 hours
5 RCS partially drained*	8 hours	2 hours	Operation not allowed	
6	24 hours	8 hours	4 hours	2 hours

* The Technical Specification 3.1.2.3 will allow operation of only ONE charging pump during this MODE and plant condition.

FOR SYSTEM 80 EXTENDED FUEL CYCLE

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3/4.1.3 MOVABLE CONTROL ASSEMBLIES

CEA POSITION

LIMITING CONDITION FOR OPERATION

3.1.3.1 All full-length (shutdown and regulating) CEAs, and all part-length CEAs which are inserted in the core, shall be OPERABLE with each CEA of a given group positioned within 6.6 inches (indicated position) of all other CEAs in its group. ~~In addition, the position of the part length CEAs Groups shall be limited to the insertion limits shown in Figure 3.1-2A.~~

APPLICABILITY: MODES 1* and 2*.

ACTION:

a. With one or more full-length CEAs inoperable due to being immovable as a result of excessive friction or mechanical interference or known to be untrippable, determine that the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied within 1 hour and be in at least HOT STANDBY within 6 hours.

b. With more than one full-length or part-length CEA inoperable or misaligned from any other CEA in its group by more than 19 inches (indicated position), be in at least HOT STANDBY within 6 hours.

c. With one or more full-length or part-length CEAs misaligned from any other CEAs in its group by more than 6.6 inches, operation in MODES 1 and 2 may continue, provided that ~~core power is reduced in accordance with Figure 3.1-2B and that~~ within 1 hour the misaligned CEA(s) is either:

1. Restored to OPERABLE status within its above specified alignment requirements, or

2. Declared inoperable and the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied. After declaring the CEA(s) inoperable, operation in MODES 1 and 2 may continue pursuant to the requirements of Specification 3.1.3.6 provided:

a) Within 1 hour the remainder of the CEAs in the group with the inoperable CEA(s) shall be aligned to within 6.6 inches of the inoperable CEA(s) while maintaining the allowable CEA sequence and insertion limits shown on Figure~~s~~ 3.1-3 ~~and 3.1-4~~; the THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.6 during subsequent operation.

*See Special Test Exceptions 3.10.2 and 3.10.4.

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and inserted beyond the Long Term Steady State Insertion Limits (Figure 3.1-5)

ACTION: (Continued)

b) The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is determined at least once per 12 hours.

Otherwise, be in at least HOT STANDBY within 6 hours.

d. With one full-length CEA inoperable due to causes other than addressed by ACTION a., above, but within its above specified alignment requirements, operation in MODES 1 and 2 may continue pursuant to the requirements of Specification 3.1.3.6.

e. With one part-length CEA inoperable and inserted in the core, operation may continue provided the alignment of the inoperable part length CEA is maintained within 6.6 inches (indicated position) of all other part-length CEAs in its group.

~~f. With part length CEAs inserted beyond insertion limits, except for surveillance testing pursuant to Specification 4.1.3.2, within 2 hours either:~~

~~1. Restore the part length CEAs to within their limits, or~~

~~2. Reduce THERMAL POWER to less than or equal to that fraction of RATED THERMAL POWER which is allowed by part length CEA group position using Figure 3.1-2A.~~

SURVEILLANCE REQUIREMENTS

4.1.3.1.1 The position of each full-length and part-length CEA shall be determined to be within 6.6 inches (indicated position) of all other CEAs in its group at least once per 12 hours except during time intervals when one CEAC is inoperable or when both CEACs are inoperable, then verify the individual CEA positions at least once per 4 hours.

4.1.3.1.2 Each full-length CEA not fully inserted and each part-length CEA which is inserted in the core shall be determined to be OPERABLE by movement of at least 5 inches in any one direction at least once per 31 days.

POSITION INDICATOR CHANNELS - OPERATING

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LIMITING CONDITION FOR OPERATION

3.1.3.2 At least two of the following three CEA position indicator channels shall be OPERABLE for each CEA:

a. CEA Reed Switch Position Transmitter (RSPT 1) with the capability of determining the absolute CEA positions within 5.2 inches,

b. CEA Reed Switch Position Transmitter (RSPT 2) with the capability of determining the absolute CEA positions within 5.2 inches, and

c. The CEA pulse counting position indicator channel.

APPLICABILITY: MODES 1 and 2.

ACTION:

With a maximum of one CEA per CEA group having only one of the above required CEA position indicator channels OPERABLE, within 6 hours either:

a. Restore the inoperable position indicator channel to OPERABLE status, or

b. Be in at least HOT STANDBY, or

c. Position the CEA group(s) with the inoperable position indicator(s) at its fully withdrawn position while maintaining the requirements of Specifications 3.1.3.1 and 3.1.3.6. Operation may then continue provided the CEA group(s) with the inoperable position indicator(s) is maintained fully withdrawn, except during surveillance testing pursuant to the requirements of Specification 4.1.3.1.2, and each CEA in the group(s) is verified fully withdrawn at least once per 12 hours thereafter by its "Full Out" limit.*

SURVEILLANCE REQUIREMENTS

4.1.3.2 Each of the above required position indicator channels shall be determined to be OPERABLE by verifying that for the same CEA, the position indicator channels agree within 5.2 inches of each other at least once per 12 hours.

*CEAs are fully withdrawn (Full Out) when withdrawn to at least 144.75 inches (193 steps).

POSITION INDICATOR CHANNELS - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.3.3 At least one CEA Reed Switch Position Transmitter indicator channel shall be OPERABLE for each shutdown, regulating or part-length CEA not fully inserted.

APPLICABILITY: MODES 3*, 4*, and 5*.

ACTION:

With less than the above required position indicator channel(s) OPERABLE, immediately open the reactor trip breakers.

SURVEILLANCE REQUIREMENTS

4.1.3.3 The above required CEA Reed Switch Position Transmitter indicator channel(s) shall be determined to be OPERABLE by performance of a CHANNEL FUNCTIONAL TEST at least once per 18 months.

*With the reactor trip breakers in the closed position.

CEA DROP TIME

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LIMITING CONDITION FOR OPERATION

3.1.3.4 The individual full-length (shutdown and regulating) CEA drop time, from a fully withdrawn position, shall be less than or equal to 4 seconds from when the electrical power is interrupted to the CEA drive mechanism until the CEA reaches its 90% insertion position with:

a. T_{cold} greater than or equal to 552°F, and

b. All reactor coolant pumps operating.

APPLICABILITY: MODES 1 and 2.

ACTION:

a. With the drop time of any full-length CEA determined to exceed the above limit, restore the CEA drop time to within the above limit prior to proceeding to MODE 1 or 2.

SURVEILLANCE REQUIREMENTS

4.1.3.4 The CEA drop time of full-length CEAs shall be demonstrated through measurement prior to reactor criticality:

a. For all CEAs following each removal and reinstallation of the reactor vessel head,

b. For specifically affected individual CEAs following any maintenance on or modification to the CEA drive system which could affect the drop time of those specific CEAs, and

c. At least once per 18 months.

SHUTDOWN CEA INSERTION LIMIT

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LIMITING CONDITION FOR OPERATION

3.1.3.5 All shutdown CEAs shall be withdrawn to at least 144.75 inches (193 steps).

APPLICABILITY: MODES 1 and 2*#.

ACTION:

With a maximum of one shutdown CEA withdrawn to less than 144.75 inches (193 steps), except for surveillance testing pursuant to Specification 4.1.3.1.2, within 1 hour either:

a. Withdraw the CEA to at least 144.75 inches (193 steps), or

b. Declare the CEA inoperable and apply Specification 3.1.3.1.

SURVEILLANCE REQUIREMENTS

4.1.3.5 Each shutdown CEA shall be determined to be withdrawn to at least 144.75 inches (193 steps):

a. Within 15 minutes prior to withdrawal of any CEAs in regulating groups during an approach to reactor criticality, and

b. At least once per 12 hours thereafter.

*See Special Test Exception 3.10.2.

#With K_{eff} greater than or equal to 1.

REGULATING CEA INSERTION LIMITS

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LIMITING CONDITION FOR OPERATION

with the

3.1.3.6 The regulating CEA groups shall be limited to the withdrawal sequence, and to the insertion limits## shown on Figure 3.1-3** ~~when the COLSS is in service or shown on Figure 3.1-4** when the COLSS is not in service~~ [redacted] CEA insertion between the Long Term Steady State Insertion Limits and the Transient Insertion Limits [redacted] restricted to:

a. Less than or equal to 4 hours per 24 hour interval,

b. Less than or equal to 5 Effective Full Power Days per 30 Effective Full Power Day interval, and

c. Less than or equal to 14 Effective Full Power Days per 18 Effective Full Power Months.

APPLICABILITY: MODES 1* and 2*#.

ACTION:

a. With the regulating CEA groups inserted beyond the Transient Insertion Limits, except for surveillance testing pursuant to Specification 4.1.3.1.2, within 2 hours either:

1. Restore the regulating CEA groups to within the limits, or

2. Reduce THERMAL POWER to less than or equal to that fraction of RATED THERMAL POWER which is allowed by the CEA group position using Figure 3.1-3 [redacted]

b. With the regulating CEA groups inserted between the Long Term Steady State Insertion Limits and the Transient Insertion Limits for intervals greater than 4 hours per 24 hour interval, operation may proceed provided either:

1. The Short Term Steady State Insertion Limits of Figure 3.1-3 [redacted] are not exceeded, or

2. Any subsequent increase in THERMAL POWER is restricted to less than or equal to 5% of RATED THERMAL POWER per hour.

*See Special Test Exceptions 3.10.2 and 3.10.4.

#With K_{eff} greater than or equal to 1.

**CEAs are fully withdrawn in accordance with Figure 3.1-3 ~~or Figure 3.1-4~~ when withdrawn to at least 144.75 inches (143 steps).

~~##Reactor power cutback will cause either (Case 1) Regulating Group 5 or Regulating Group 4 and 5 to be dropped with no sequential insertion of additional Regulating Groups (Groups 1, 2, 3, and 4) or (Case 2) Regulating Group 5 or Regulating Group 4 and 5 to be dropped with all or part of the remaining Regulating Groups (Groups 1, 2, 3, and 4) being sequentially inserted. In either case, the Transient Insertion Limit and the withdrawal sequence of Figure 3.1-3 [redacted] can be exceeded for up to 2 hours.~~

Insert

Insert

Following a reactor power cutback in which (1) Regulating Group 5 is dropped or (2) Regulating Groups 4 and 5 are dropped and for cases (1) and (2) should the remaining Regulating Groups (Group 1, 2, 3, and 4) be sequentially inserted, the Transient Insertion Limit of Figure 3.1-3 can be exceeded for up to 2 hours. Also for cases (1) and (2), the specified overlap between Regulating Groups 3, 4 and 5 can be exceeded for up to 2 hours.

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ACTION: (Continued)

c. With the regulating CEA groups inserted between the Long Term Steady State Insertion Limits and the Transient Insertion Limits for intervals greater than 5 EFPD per 30 EFPD interval or greater than 14 EFPD per 18 Effective Full Power Months, either:

   1. Restore the regulating groups to within the Long Term Steady State Insertion Limits within 2 hours, or

   2. Be in at least HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.6 The position of each regulating CEA group shall be determined to be within the Transient Insertion Limits at least once per 12 hours except during time intervals when the PDIL Auctioneer Alarm Circuit is inoperable, then verify the individual CEA positions at least once per 4 hours. The accumulated times during which the regulating CEA groups are inserted beyond the Long Term Steady State Insertion Limits but within the Transient Insertion Limits shall be determined at least once per 24 hours.

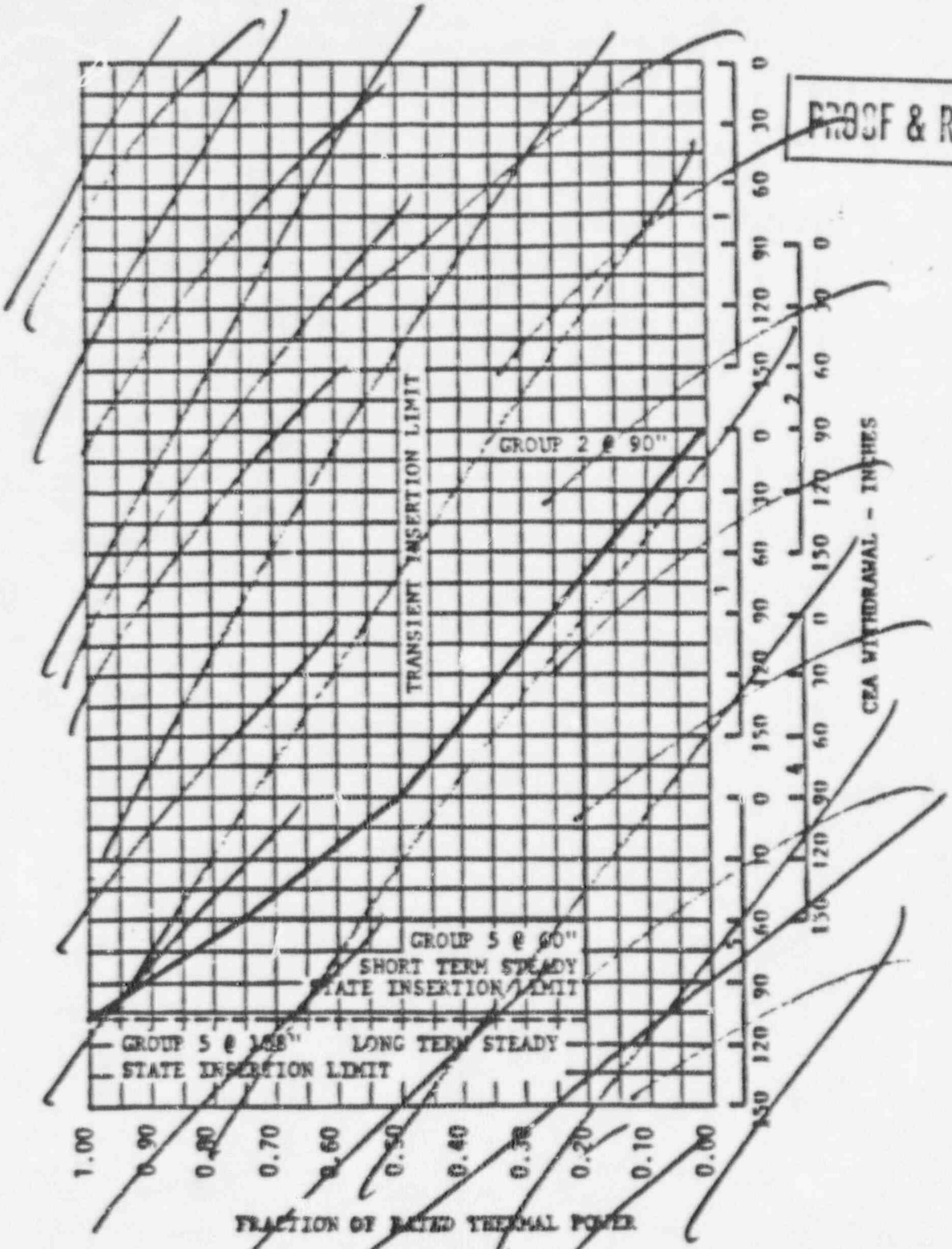

FIGURE 3.1-3

CEA INSERTION LIMITS VS THERMAL POWER
~~(COLSS IN SERVICE)~~ ← SEE APLICANT'S SAR

3/4 2.1 LINEAR HEAT RATE

LIMITING CONDITION FOR OPERATION

3.2.1 The linear heat rate shall not exceed 14.0 kW/ft.

APPLICABILITY: MODE 1 above 20% of RATED THERMAL POWER.

ACTION:

With the linear heat rate exceeding its limits, as indicated by either (1) the COLSS calculated core power exceeding the COLSS calculated core power operating limit based on kW/ft; or (2) when the COLSS is not being used, any OPERABLE Local Power Density channel exceeding the linear heat rate limit, within 15 minutes initiate corrective action to reduce the linear heat rate to within the limits and either:

a. Restore the linear heat rate to within its limits within 1 hour, or

b. Reduce THERMAL POWER to less than or equal to 20% of RATED THERMAL POWER within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.2.1.1 The provisions of Specification 4.0.4 are not applicable.

4.2.1.2 The linear heat rate shall be determined to be within its limits when THERMAL POWER is above 20% of RATED THERMAL POWER by continuously monitoring the core power distribution with the Core Operating Limit Supervisory System (COLSS) or, with the COLSS out of service, by verifying at least once per 2 hours that the linear heat rate, as indicated on all OPERABLE Local Power Density channels, is less than or equal to 14.0 kW/ft.

4.2.1.3 At least once per 31 days, the COLSS Margin Alarm shall be verified to actuate at a THERMAL POWER level less than or equal to the core power operating limit based on 14.0 kW/ft.

3/4.2.2 PLANAR RADIAL PEAKING FACTORS - F_{xy}

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LIMITING CONDITION FOR OPERATION

3.2.2 The measured PLANAR RADIAL PEAKING FACTORS (F^m_{xy}) shall be less than or equal to the PLANAR RADIAL PEAKING FACTORS (F^c_{xy}) used in the Core Operating Limit Supervisory System (COLSS) and in the Core Protection Calculators (CPC).

APPLICABILITY: MODE 1 above 20% of RATED THERMAL POWER.*

ACTION:

With an F^m_{xy} exceeding a corresponding F^c_{xy}, within 6 hours either:

a. Adjust the CPC addressable constants to increase the multiplier applied to planar radial peaking by a factor equivalent to greater than or equal to F^m_{xy}/F^c_{xy} and restrict subsequent operation so that a margin to the COLSS operating limits of at least $[(F^m_{xy}/F^c_{xy}) - 1.0] \times 100\%$ is maintained; or

b. Adjust the affected PLANAR RADIAL PEAKING FACTORS (F^c_{xy}) used in the COLSS and CPC to a value greater than or equal to the measured PLANAR RADIAL PEAKING FACTORS (F^m_{xy}) or

c. Reduce THERMAL POWER to less than or equal to 20% of RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS

4.2.2.1 The provisions of Specification 4.0.4 are not applicable.

4.2.2.2 The measured PLANAR RADIAL PEAKING FACTORS (F^m_{xy}) obtained by using the incore detection system, shall be determined to be less than or equal to the PLANAR RADIAL PEAKING FACTORS (F^c_{xy}), used in the COLSS and CPC at the following intervals:

a. After each fuel loading with THERMAL POWER greater than 40% but prior to operation above 70% of RATED THERMAL POWER, and

b. At least once per 31 Effective Full Power Days.

*See Special Test Exception 3.10.2.

3/4.2.3 AZIMUTHAL POWER TILT - T_q

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LIMITING CONDITION FOR OPERATION

3.2.3 The AZIMUTHAL POWER TILT (T_q) shall be less than or equal to the AZIMUTHAL POWER TILT Allowance used in the Core Protection Calculators (CPCs).

APPLICABILITY: MODE 1 above 20% of RATED THERMAL POWER.*

ACTION:

a. With the measured AZIMUTHAL POWER TILT determined to exceed the AZIMUTHAL POWER TILT Allowance used in the CPCs but less than or equal to 0.10, within 2 hours either correct the power tilt or adjust the AZIMUTHAL POWER TILT Allowance used in the CPCs to greater than or equal to the measured value.

b. With the measured AZIMUTHAL POWER TILT determined to exceed 0.10:

1. Due to misalignment of either a part-length or full-length CEA, within 30 minutes verify that the Core Operating Limit Supervisory System (COLSS) (when COLSS is being used to monitor the core power distribution per Specifications 4.2.1 and 4.2.4) is detecting the CEA misalignment.

2. Verify that the AZIMUTHAL POWER TILT is within its limit within 2 hours after exceeding the limit or reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within the next 2 hours and verify that the Variable Overpower Trip Setpoint has been reduced as appropriate within the next 4 hours.

3. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above 50% of RATED THERMAL POWER may proceed provided that the AZIMUTHAL POWER TILT is verified within its limit at least once per hour for 12 hours or until verified acceptable at 95% or greater RATED THERMAL POWER.

*See Special Test Exception 3.10.2.

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SURVEILLANCE REQUIREMENTS

4.2.3.1 The provisions of Specification 4.0.4 are not applicable.

4.2.3.2 The AZIMUTHAL POWER TILT shall be determined to be within the limit above 20% of RATED THERMAL POWER by:

a. Continuously monitoring the tilt with COLSS when the COLSS is OPERABLE.

b. Calculating the tilt at least once per 12 hours when the COLSS is inoperable.

c. Verifying at least once per 31 days, that the COLSS Azimuthal Tilt Alarm is actuated at an AZIMUTHAL POWER TILT ~~less than or equal~~ greater than to the AZIMUTHAL POWER TILT Allowance used in the CPCs.

d. Using the incore detectors at least once per 31 EFPD to independently confirm the validity of the COLSS calculated AZIMUTHAL POWER TILT.

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3/4.2.4 DNBR MARGIN

LIMITING CONDITION FOR OPERATION

3.2.4 The DNBR margin shall be maintained by operating within the Region of Acceptable Operation of Figure 3.2-1 or 3.2-2, as applicable. ~~or in accordance with the requirements of Action 6 of Table 3.3-1.~~

APPLICABILITY: MODE 1 above 20% of RATED THERMAL POWER.

ACTION:

With operation outside of the region of acceptable operation, as indicated by either (1) the COLSS calculated core power exceeding the COLSS calculated core power operating limit based on DNBR; or (2) when the COLSS is not being used, any OPERABLE Low DNBR channel below the DNBR limit, within 15 minutes initiate corrective action to restore either the DNBR core power operating limit or the DNBR margin to within the limits and either:

a. Restore the DNBR core power operating limit or DNBR margin to within its limits within 1 hour, or

b. Reduce THERMAL POWER to less than or equal to 20% of RATED THERMAL POWER within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.2.4.1 The provisions of Specification 4.0.4 are not applicable.

4.2.4.2 The DNBR shall be determined to be within its limits when THERMAL POWER is above 20% of RATED THERMAL POWER by continuously monitoring the core power distribution with the Core Operating Limit Supervisory System (COLSS) or, with the COLSS out of service, by verifying at least once per 2 hours that the DNBR margin, as indicated on all OPERABLE DNBR margin channels, is within the limit shown on Figure 3.2-2.

4.2.4.3 At least once per 31 days, the COLSS Margin Alarm shall be verified to actuate at a THERMAL POWER level less than or equal to the core power operating limit based on DNBR.

4.2.4.4 The following DNBR or equivalent penalty factors shall be verified to be included in the COLSS and CPC DNBR calculations at least once per 31 EFPD.

Burnup (GWD/MTU)	DNBR Penalty (%)*
0-10	0.5
10-20	1.0
20-30	2.0
30-40	3.5
40-50	5.5

*The penalty for each batch will be determined from the batch's maximum burnup assembly and applied to the batch's maximum radial power peak assembly. A single net penalty for COLSS and CPC will be determined from the penalties associated with each batch accounting for the offsetting margins due to the lower radial power peaks in the higher burnup batches.

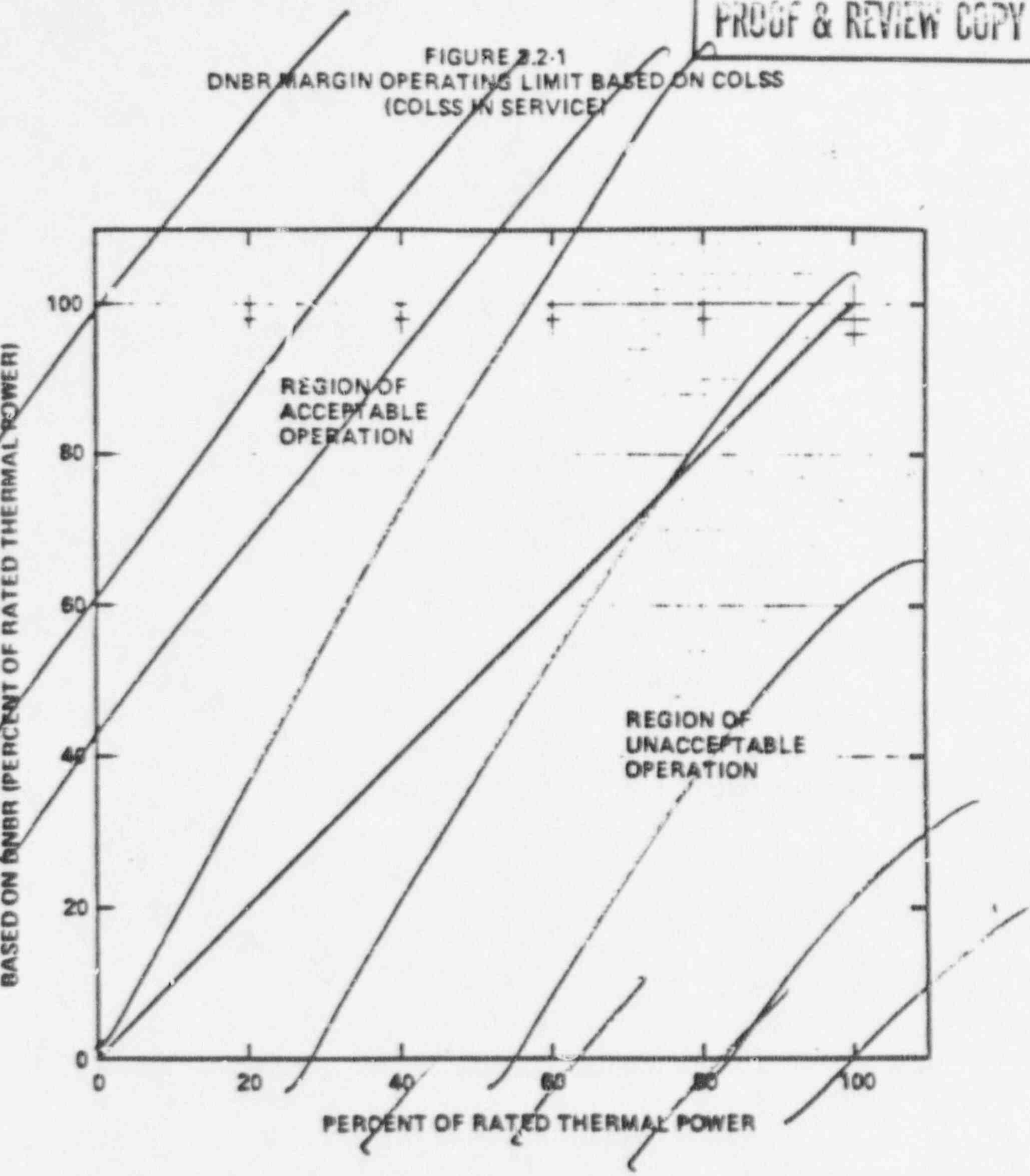
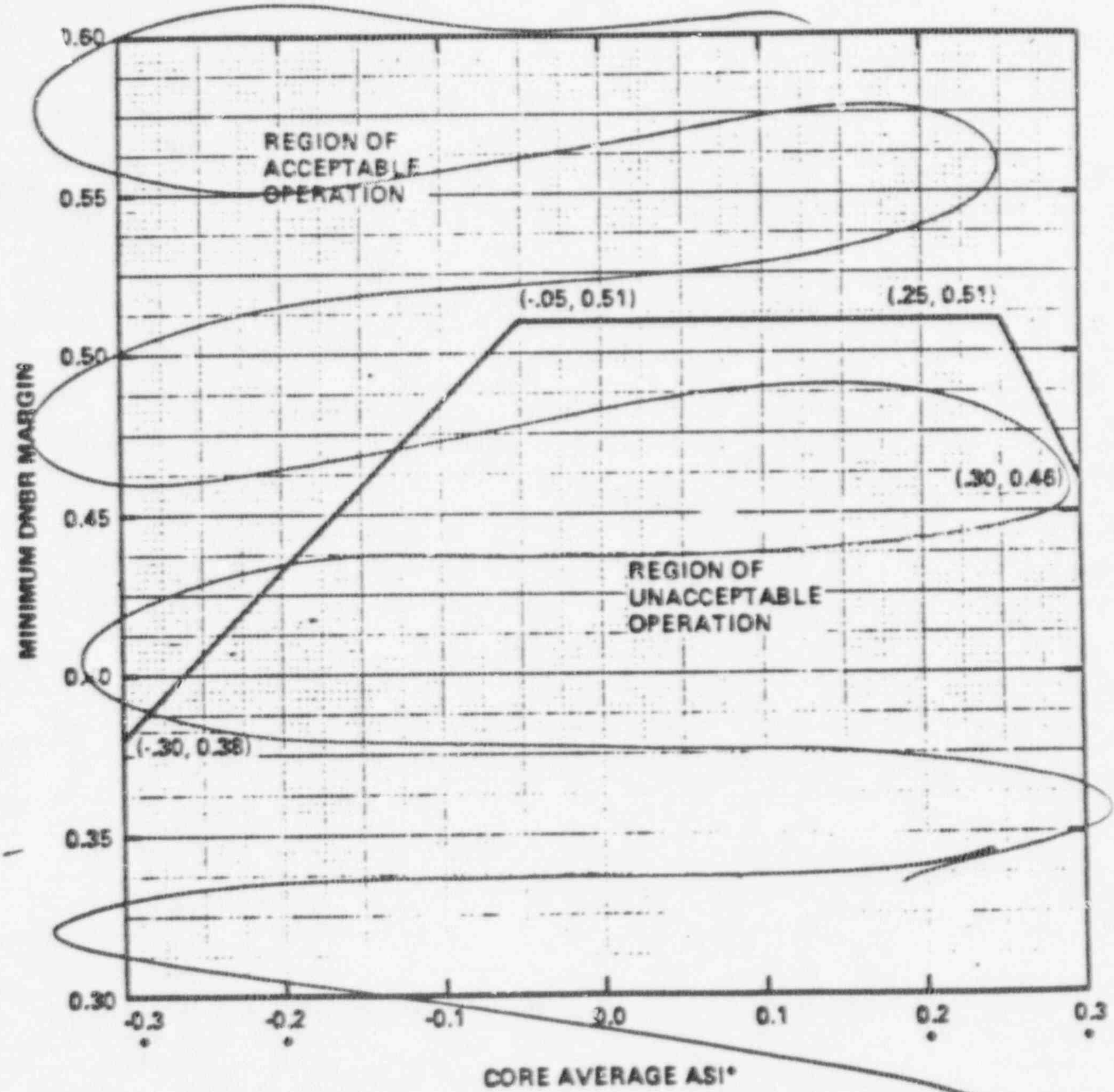


FIGURE 3.2-1

DNBR MARGIN OPERATING LIMIT BASED ON COLSS
(COLSS IN SERVICE)

SEE APPLICANT'S SAR

~~FIGURE 3.2-2~~
~~DNBR MARGIN OPERATING LIMIT BASED ON CORE PROTECTION CALCULATOR~~
~~(COLSS OUT OF SERVICE)~~



~~*SEE SECTION 3.2.7 FOR THE ASI OPERATING LIMITS~~

FIGURE 3.2-2

DNBR MARGIN OPERATING LIMIT BASED ON CORE PROTECTION CALCULATORS

(COLSS OUT OF SERVICE)

SEE APPLICANT'S SAR

3/4.2.5 RCS FLOW RATE

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LIMITING CONDITION FOR OPERATION

3.2.5 The actual Reactor Coolant System total flow rate shall be greater than or equal to 164.0 x 10^6 lbm/hr.

APPLICABILITY: MODE 1.

ACTION:

With the actual Reactor Coolant System total flow rate determined to be less than the above limit, reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.5 The actual Reactor Coolant System total flow rate shall be determined to be greater than or equal to its limit at least once per 12 hours.

3/4.2.6 REACTOR COOLANT COLD LEG TEMPERATURE

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LIMITING CONDITION FOR OPERATION

3.2.6 The reactor coolant cold leg temperature (T_c) shall be within the Area of Acceptable Operation shown in Figure 3.2-3.

APPLICABILITY: MODE 1* and 2*

ACTION:

With the reactor coolant cold leg temperature exceeding its limit, restore the temperature to within its limit within 2 hours or be in HOT STANDBY within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.2.6 The reactor coolant cold leg temperature shall be determined to be within its limit at least once per 12 hours.

*See Special Test Exception 3.10.4.

~~#With K_{eff} greater than or equal to 1.~~

FIGURE 3.2-3
REACTOR COOLANT COLD LEG TEMPERATURE vs CORE POWER LEVEL

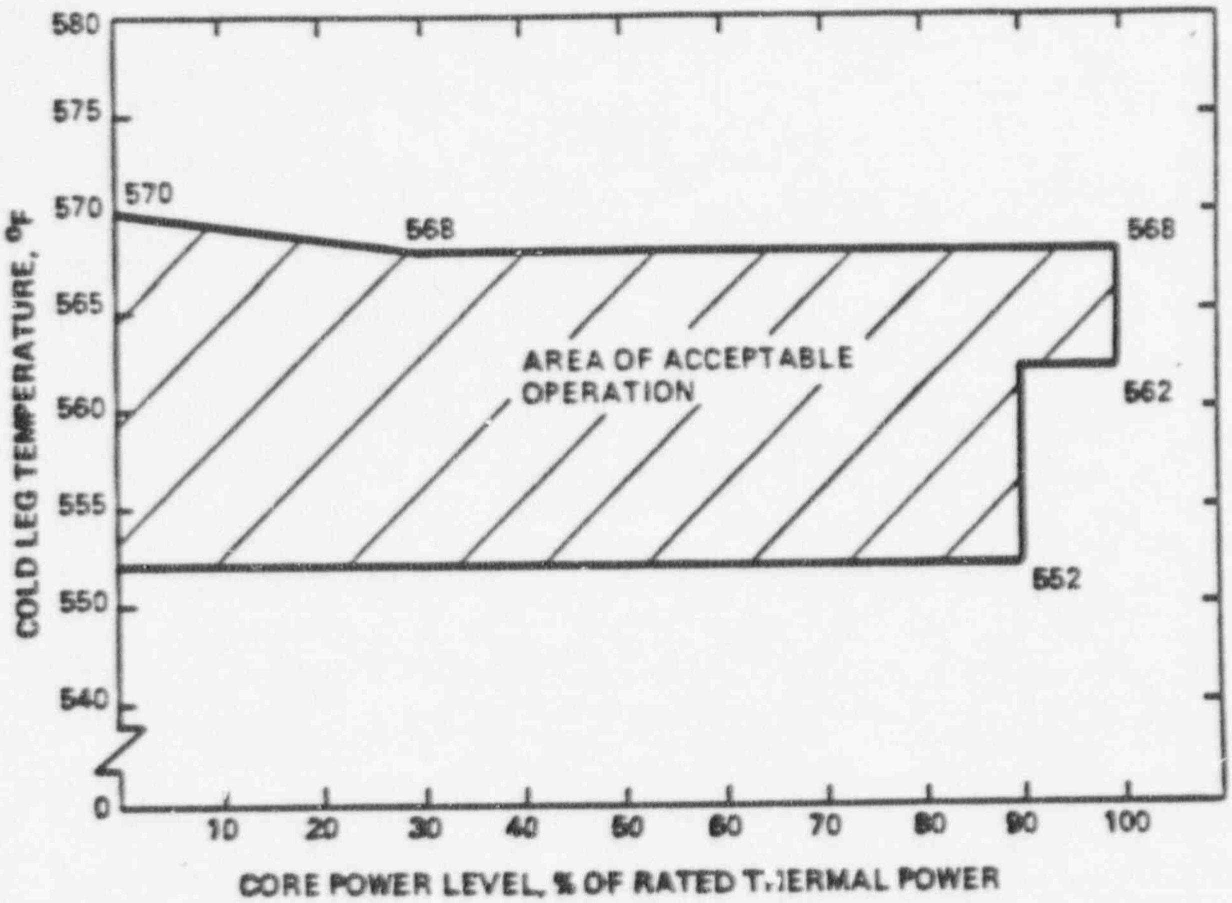

FIGURE 3.2-3

REACTOR COOLANT COLD LEG TEMPERATURE VS CORE POWER LEVEL

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3/4.2.7 AXIAL SHAPE INDEX

LIMITING CONDITION FOR OPERATION

3.2.7 The core average AXIAL SHAPE INDEX (ASI) shall be maintained within the following limits:

a. COLSS OPERABLE
$-0.28 \leq \text{ASI} \leq 0.28$

b. COLSS OUT OF SERVICE (CPC)
$-0.20 \leq \text{ASI} \leq +0.20$

APPLICABILITY: MODE 1 above 20% of RATED THERMAL POWER*.

ACTION:

With the core average AXIAL SHAPE INDEX outside its above limits, restore the core average ASI to within its limit within 2 hours or reduce THERMAL POWER to less than 20% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.7 The core average AXIAL SHAPE INDEX shall be determined to be within its limit at least once per 12 hours using the COLSS or any OPERABLE Core Protection Calculator channel.

*See Special Test Exception 3.10.2.

3/4.2.8 PRESSURIZER PRESSURE

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LIMITING CONDITION FOR OPERATION

3.2.8 The pressurizer pressure shall be maintained between 1815 psia and 2370 psia.

APPLICABILITY: MODES 1 and 2.*

ACTION:

With the pressurizer pressure outside its above limits, restore the pressure to within its limit within 2 hours or be in at least HOT STANDBY within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.2.8 The pressurizer pressure shall be determined to be within its limit at least once per 12 hours.

*See Special Test Exception 3.10.5

3/4.3.1 REACTOR PROTECTIVE INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.1 As a minimum, the reactor protective instrumentation channels and bypasses of Table 3.3-1 shall be OPERABLE with RESPONSE TIMES as shown in Table 3.3-2.

APPLICABILITY: As shown in Table 3.3-1.

ACTION:

As shown in Table 3.3-1.

SURVEILLANCE REQUIREMENTS

4.3.1.1 Each reactor protective instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations for the MODES and at the frequencies shown in Table 4.3-1.

4.3.1.2 The logic for the bypasses shall be demonstrated OPERABLE prior to each reactor startup unless performed during the preceding 92 days. The total bypass function shall be demonstrated OPERABLE at least once per 18 months during CHANNEL CALIBRATION testing of each channel affected by bypass operation.

4.3.1.3 The REACTOR TRIP SYSTEM RESPONSE TIME of each reactor trip function shall be demonstrated to be within its limit at least once per 18 months. Each test shall include at least one channel per function such that all channels are tested at least once every N times 18 months where N is the total number of redundant channels in a specific reactor trip function as shown in the "Total No. of Channels" column of Table 3.3-1.

4.3.1.4 The isolation characteristics of each CEA isolation amplifier shall be verified at least once per 18 months during the shutdown per the following tests for the CEA position isolation amplifiers:

a. With 120 volts A.C. (60 Hz) applied for at least 30 seconds across the output, the reading on the input does not change by more than 0.015 volt D.C. with an applied input voltage of 5-10 volts D.C.

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SURVEILLANCE REQUIREMENTS (Continued)

b. With 120 volts A.C. (60 Hz) applied for at least 30 seconds across the input, the reading on the output does not exceed 15 volts D.C.

4.3.1.5 The Core Protection Calculators shall be determined OPERABLE at least once per 12 hours by verifying that less than three auto restarts have occurred on each calculator during the past 12 hours. The auto restart periodic tests Restart (Code 30) and Normal System Load (Code 33) shall not be included in this total.

4.3.1.6 The Core Protection Calculators shall be subjected to a CHANNEL FUNCTIONAL TEST to verify OPERABILITY within 12 hours of receipt of a High CPC Cabinet Temperature alarm.

TABLE 3.3-1

REACTOR PROTECTIVE INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
I. TRIP GENERATION					
A. Process					
1. Pressurizer Pressure - High	4	2	3	1, 2	2#, 3#
2. Pressurizer Pressure - Low	4	2 (b)	3	1, 2	2#, 3#
3. Steam Generator Level - Low	4/SG	2/SG	3/SG	1, 2	2#, 3#
4. Steam Generator Level - High	4/SG	2/SG	3/SG	1, 2	2#, 3#
5. Steam Generator Pressure - Low	4/SG	2/SG	3/SG	1, 2, 3*, 4*	2#, 3#
6. Containment Pressure - High	4	2	3	1, 2	2#, 3#
7. Reactor Coolant Flow - Low	4/SG	2/SG	3/SG	1, 2	2#, 3#
8. Local Power Density - High	4	2 (c)(d)	3	1, 2	2#, 3#
9. DNBR - Low	4	2 (c)(d)	3	1, 2	2#, 3#
10. Linear Power Level - High	**	**	**	**	** (B
11. Loss of Load	**	**	**	**	** (B
B. Excore Neutron Flux					
1. Variable Overpower Trip	4	2	3	1, 2	2#, 3#
2. Logarithmic Power Level - High					
a. Startup and Operating	4	2 (a)(d)	3	1, 2	2#, 3#
	4	2	3	3*, 4*, 5*	8
b. Shutdown	4	0	2	3, 4, 5	4
C. Core Protection Calculator System					
1. CEA Calculators	2	1	2 (e)	1, 2	6, 7
2. Core Protection Calculators	4	2 (c)(d)	3	1, 2	2#, 3#, 7

CESSAR 80-NSSS-STS

3/4 3-4

TABLE 3.3-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
D. Supplementary Protection System Pressurizer Pressure - High	4 (f)	2	3	1, 2	8
II. RPS LOGIC					
A. Matrix Logic	6	1	3	1, 2	1
	6	1	3	3*, 4*, 5*	8
B. Initiation Logic	4	2	4	1, 2	5
	4	2	4	3*, 4*, 5*	8
III. RPS ACTUATION DEVICES					
A. Reactor Trip Breaker	4 (f)	2	4	1, 2	5
	4 (f)	2	4	3*, 4*, 5*	8
B. Manual Trip	4 (f)	2	4	1, 2	5
	4 (f)	2	4	3*, 4*, 5*	8

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TABLE 3.3-1 (Continued)

TABLE NOTATIONS

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*With the protective system trip breakers in the closed position, the CEA drive system capable of CEA withdrawal, and fuel in the reactor vessel.

#The provisions of Specification 3.0.4 are not applicable.

(a) Trip may be manually bypassed above 10^{-4}% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is less than or equal to 10^{-4}% of RATED THERMAL POWER.

(b) Trip may be manually bypassed below 400 psia; bypass shall be automatically removed whenever pressurizer pressure is greater than or equal to 500 psia.

(c) Trip may be manually bypassed below 1% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is greater than or equal to 1% of RATED THERMAL POWER.

(d) Trip may be bypassed during testing pursuant to Special Test Exception 3.10.3.

(e) See Special Test Exception 3.10.2.

(f) There are four channels, each of which is comprised of one of the four reactor trip breakers, arranged in a selective two-out-of-four configuration (i.e., one-out-of-two taken twice)

ACTION STATEMENTS

ACTION 1 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and/or open the protective system trip breakers.

ACTION 2 - With the number of channels OPERABLE one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may continue provided the inoperable channel is placed in the bypassed or tripped condition within 1 hour. If the inoperable channel is bypassed, the desirability of maintaining this channel in the bypassed condition shall be reviewed in accordance with Specification 6.5.1.6.h of Administrative Controls. The channel shall be returned to OPERABLE status no later than during the next COLD SHUTDOWN.

ACTION STATEMENTS

With a channel process measurement circuit that affects multiple functional units inoperable or in test, bypass or trip all associated functional units as listed below:

Process Measurement Circuit	Functional Units Bypassed/Tripped
1. Excore Nuclear Instrument-Linear Power (Subchannel or Linear)	Variable Overpower (RPS) Local Power Density - High (RPS) DNBR - Low (RPS)
2. Pressurizer Pressure - High (Narrow Range)	Pressurizer Pressure - High (RPS) Local Power Density - High (RPS) DNBR - Low (RPS)
3. Steam Generator Pressure - Low	Steam Generator Pressure - Low Steam Generator Level 1-Low (ESF) Steam Generator Level 2-Low (ESF)
4. Steam Generator Level - Low (Wide Range)	Steam Generator Level - Low (RPS) Steam Generator Level 1-Low (ESF) Steam Generator Level 2-Low (ESF)
5. Core Protection Calculator	Local Power Density - High (RPS) DNBR - Low (RPS)

ACTION 3 - With the number of channels OPERABLE one less than the Minimum Channels OPERABLE requirement, STARTUP and/or POWER OPERATION may continue provided the following conditions are satisfied:

a. Verify that one of the inoperable channels has been bypassed and place the other channel in the tripped condition within 1 hour, and

b. All functional units affected by the bypassed/tripped channel shall also be placed in the bypassed/tripped condition as listed below:

Process Measurement Circuit	Functional Unit Bypassed/Tripped
1. Excore Nuclear Instrument-Linear Power (Subchannel or Linear)	Variable Overpower (RPS) Local Power Density - High (RPS) DNBR - Low (RPS)
2. Pressurizer Pressure - High (Narrow Range)	Pressurizer Pressure - High (RPS) Local Power Density - High (RPS) DNBR - Low (RPS)

ACTION STATEMENTS

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3.	Steam Generator Pressure - Low	Steam Generator Pressure - Low Steam Generator Level 1-Low (ESF) Steam Generator Level 2-Low (ESF)
4.	Steam Generator Level - Low (Wide Range)	Steam Generator Level - Low (RPS) Steam Generator Level 1-Low (ESF) Steam Generator Level 2-Low (ESF)
5.	Core Protection Calculator	Local Power Density - High (RPS) DNBR - Low (RPS)

STARTUP and/or POWER OPERATION may continue until the performance of the next required CHANNEL FUNCTIONAL TEST. Subsequent STARTUP and/or POWER OPERATION may continue if one channel is restored to OPERABLE status and the provisions of ACTION 2 are satisfied.

ACTION 4 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, suspend all operations involving positive reactivity changes.

ACTION 5 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, STARTUP and/or POWER OPERATION may continue provided the reactor trip breaker of the inoperable channel is placed in the tripped condition within 1 hour, otherwise, be in at least HOT STANDBY within 6 hours; however, the trip breaker associated with the inoperable channel may be closed for up to 1 hour for surveillance testing per Specification 4.3.1.1.

ACTION 6 - a. With one CEAC inoperable, operation may continue for up to 7 days provided that at least once per 4 hours, each CEA is verified to be within 6.6 inches (indicated position) of all other CEAs in its group. ~~After 7 days, operation may continue provided that the conditions of Action Item 6.b or 6.c are met.~~

b. With both CEACs inoperable ~~and COLSS in service~~, operation may continue provided that:

INSERT

1. Within 1 hour:

a) Operation is restricted to the limits shown in Figure 3.3-1. The DNBR margin required by Specification 3.2.4 is replaced by this restriction when both CEAC's are inoperable and COLSS is in operation.

b) The Linear Heat Rate Margin required by Specification 3.2.1 is maintained.

c) The Reactor Power Cutback System is placed out of service.

INSERT

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1. Within 1 hour the margins required by Specifications 3.2.1 and 3.2.4 are increased and maintained at a value equivalent to greater than or equal to 19% of RATED THERMAL POWER.

ACTION STATEMENTS

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2. Within 4 hours:

a) All full-length and part-length CEA groups are withdrawn to and subsequently maintained at the "Full Out" position, except during surveillance testing pursuant to the requirements of Specification 4.1.3.1.2 or for control when CEA group 5 may be inserted no further than 127.5 inches withdrawn.

b) The "RSPT/CEAC Inoperable" the addressable constant in the CPCs is set to ~~be indicated that both CEACs are~~ inoperable status.

c) The Control Element Drive Mechanism Control System (CEDMCS) is placed in and subsequently maintained in the "Standby" mode except during CEA group 5 motion permitted by a) above, when the CEDMCS may be operated in either the "Manual Group" or "Manual Individual" mode.

3. At least once per 4 hours, all full-length and part-length CEAs are verified fully withdrawn except during surveillance testing pursuant to Specification 4.1.3.1.2 or during insertion of CEA group 5 as permitted by 2.a) above, then verify at least once per 4 hours that the inserted CEAs are aligned within 6.6 inches (indicated position) of all other CEAs in its group.

4. Following a CEA misalignment with both CEAC's inoperable and COLSS in operation, operation may continue provided that within 1 hour:

The power is reduced to 85% of the pre-misaligned power but need not be reduced to less than 50% of RATED THERMAL POWER. This power restriction replaces the power restriction of Specification 3.1.3.1, otherwise Specification 3.1.3.1 remains applicable.

c. With both CEACs inoperable and COLSS out-of-service, operation may continue provided that:

1. Within 1 hour:

a) The existing CPC value of the CPC addressable constant "BERR1" is multipled by 1.19 and the resulting value is re-entered into the CPCs.

b) The Reactor Power Cutback System is placed out of service

c) The COLSS out of service Limit Line, Specification 3.2.4, is not applicable to this mode of operation.

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ACTION STATEMENTS

2. Within 4 hours:

a) All full length and part length CEA groups are withdrawn to and subsequently maintained at the "Full Out" position, except during surveillance testing pursuant to the requirements of Specification 4.1.3.1.2 or for control when CEA group 5 may be inserted no further than 127.5 inches withdrawn.

b) The "RSPT/CEAC Inoperable" addressable constant in the CPCs is set to be indicated that both CEAC's are inoperable.

c) The Control Element Drive Mechanism Control System (CEDMCS) is placed in and subsequently maintained in the "Standby" mode except during CEA group 5 motion permitted by a) above, when the CEDMCS may be operated in either the "Manual Group" or "Manual Individual" mode.

3. At least once per 4 hours, all full length and part length CEAs are verified fully withdrawn except during surveillance testing pursuant to Specification 4.1.3.1.2 or during insertion of CEA group 5 as permitted by 2.a) above, then verify at least once per 4 hours that the inserted CEAs are aligned within 6.6 inches (indicated position) of all other CEAs in its group.

4. Following a CEA misalignment with both CEAC's and COLSS inoperable, operation may continue provided that within 1 hour:

The power is reduced to 85% of the pre misaligned power but need not be reduced to less than 50% of RATED THERMAL POWER. This power restriction replaces the power restriction of Specification 3.1.3.1, otherwise Specification 3.1.3.1 remains applicable.

ACTION 7 - With three or more auto restarts, excluding periodic auto restarts (Code 30 and Code 33), of one non-bypassed calculator during a 12-hour interval, demonstrate calculator OPERABILITY by performing a CHANNEL FUNCTIONAL TEST within the next 24 hours.

ACTION 8 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore an inoperable channel to OPERABLE status within 48 hours or open an affected reactor trip breaker within the next hour.

CESSAR80-NSSS-STS

3/4 3-7g

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TABLE 3.3-2

REACTOR PROTECTIVE INSTRUMENTATION RESPONSE TIMES 1

FUNCTIONAL UNIT	RESPONSE TIME
I. TRIP GENERATION	
A. Process	
1. Pressurizer Pressure - High	≤ [redacted] seconds
2. Pressurizer Pressure - Low	≤ [redacted] seconds
3. Steam Generator Level - Low	≤ [redacted] seconds
4. Steam Generator Level - High	≤ [redacted] seconds
5. Steam Generator Pressure - Low	≤ [redacted] seconds
6. Containment Pressure - High	≤ [redacted] seconds
7. Reactor Coolant Flow - Low	≤ [redacted] second
8. Local Power Density - High	
a. Neutron Flux Power from Excore Neutron Detectors	≤ [redacted] second*
b. CEA Positions	≤ [redacted] second**
c. CEA Positions: CEAC Penalty Factor	≤ [redacted] second**
9. DNBR - Low***	
a. Neutron Flux Power from Excore Neutron Detectors	≤ [redacted] second*
b. CEA Positions	≤ [redacted] second**
c. Cold Leg Temperature	≤ [redacted] second##
d. Hot Leg Temperature	≤ [redacted] second##
e. Primary Coolant Pump Shaft Speed	≤ [redacted] second#
f. Reactor Coolant Pressure from Pressurizer	≤ [redacted] second###
g. CEA Positions: CEAC Penalty Factor	≤ [redacted] second**
10. Linear Power Level	(BOP)
11. Loss of Load	(BOP)
B. Excore Neutron Flux	
1. Variable Overpower Trip	≤ [redacted] second*
2. Logarithmic Power Level - High	
a. Startup and Operating	≤ [redacted] second*
b. Shutdown	≤ [redacted] second*

1 See Applicants SAR

CESSAR-NSSS-STS

TABLE 3.3-2 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION RESPONSE TIMES 1

FUNCTIONAL UNIT	RESPONSE TIME
C. Core Protection Calculator System	
1. CEA Calculators	Not Applicable
2. Core Protection Calculators	Not Applicable
D. Supplementary Protection System	
Pressurizer Pressure - High	≤ [redacted] second
II. RPS LOGIC	
A. Matrix Logic	Not Applicable
B. Initiation Logic	Not Applicable
III. RPS ACTUATION DEVICES	
A. Reactor Trip Breakers	Not Applicable
B. Manual Trip	Not Applicable

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*Neutron detectors are exempt from response time testing. The response time of the neutron flux signal portion of the channel shall be measured from the detector output or from the input of first electronic component in channel.

**~~Response time shall be measured from the output of the sensor. Acceptable CEA sensor response shall be demonstrated by compliance with Specification 3.1.3.4.~~

*** Insert

#The pulse transmitters measuring pump speed are exempt from response time testing. The response time shall be measured from the pulse shaper input.

##Response time shall be measured from the output of the resistance temperature detector (sensor). RTD response time shall be measured at least once per 18 months. The measured response time of the slowest RTD shall be less than or equal to ~~[illegible]~~ 6 seconds. ~~Adjustments to the CPC addressable constants given in Table 3.3-2a shall be made to accomodate current values of the RTD time constants. If the RTD time constant for a CPC channel exceeds the value corresponding to the penalties currently in use, the affected channel(s) shall be declared inoperable until penalties appropriate to the new time constant are installed.~~

###Response time shall be measured from the output of the pressure transmitter. The transmitter response time shall be less than or equal to 0.7 second.

Insert

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** The CEA position transmitters are exempt from response time testing. The response time shall be measured from the input to the CPC, CEAC or signal isolator.

*** Response times are verified using CPC Response Time Test Software, and are for hardware delays only.

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TABLE 3.3-2a

INCREASES IN BERR0, BERR2, AND BERR4 VERSUS RTD DELAY TIMES

RTD DELAY TIME (τ)	BERR0 INCREASE (%)	BERR2 INCREASE (%)	BERR4 INCREASE (%)
$\tau \leq 8.0$ sec	0	0	0
8.0 sec $< \tau \leq 10.0$ sec	2.5	2.0	1.0
10.0 sec $< \tau \leq 13.0$ sec	6.0	4.0	6.0

NOTE: BERR term increases are not cumulative. For example, if the time constant changes from the range of $8.0 < \tau \leq 10.0$ sec to the range $10.0 < \tau \leq 13.0$, the BERR0 increase from its original ($\tau \leq 8.0$ sec) value is 6.0 not 2.5 + 6.0.

TABLE 4.3-1

REACTOR PROTECTIVE INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
I. TRIP GENERATION				
A. Process				
1. Pressurizer Pressure - High	S	R	M	1, 2
2. Pressurizer Pressure - Low	S	R	M	1, 2
3. Steam Generator Level - Low	S	R	M	1, 2
4. Steam Generator Level - High	S	R	M	1, 2
5. Steam Generator Pressure - Low	S	R	M	1, 2, 3*, 4*
6. Containment Pressure - High	S	R	M	1, 2
7. Reactor Coolant Flow - Low	S	R	M	1, 2
8. Local Power Density - High	S	~~D (2, 4), R (4, 5)~~	~~M, R (6)~~	1, 2
9. DNBR - Low	S	~~D (2, 4), R (4, 5) M (8), S (7)~~	~~M, R (6)~~	1, 2
10. Linear Power Level - High	**	**	**	** (BOP)
11. Loss of Load	**	**	**	** (BOP)
B. Excore Neutron Flux				
1. Variable Overpower Trip	S	D (2, 4), M (3, 4) Q (4)	M	1, 2
2. Logarithmic Power Level - High	S	R (4)	M and S/U (1)	1, 2, 3, 4, 5 and *
C. Core Protection Calculator System				
1. CEA Calculators	S	R	M, R (6)	1, 2
2. Core Protection Calculators	S	D (2, 4), R (4, 5) M (8), S (7)	M (9), R (6)	1, 2

See Core Protection Calculation System

** See Applicant SAR

TABLE 4.3-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
D. Supplementary Protection System				
Pressurizer Pressure - High	S	R	M	1, 2
II. RPS LOGIC				
A. Matrix Logic	N.A.	N.A.	M	1, 2, 3*, 4*, 5*
B. Initiation Logic	N.A.	N.A.	M	1, 2, 3*, 4*, 5*
III. RPS ACTUATION DEVICES				
A. Reactor Trip Breakers	N.A.	N.A.	M, R (10)	1, 2, 3*, 4*, 5*
B. Manual Trip	N.A.	N.A.	M	1, 2, 3*, 4*, 5*

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TABLE 4.3-1 (Continued)

TABLE NOTATIONS

* - With reactor trip breakers in the closed position and the CEA drive system capable of CEA withdrawal, and fuel in the reactor vessel.

(1) - Each STARTUP or when required with the reactor trip breakers closed and the CEA drive system capable of rod withdrawal, if not performed in the previous 7 days.

(2) - Heat balance only (CHANNEL FUNCTIONAL TEST not included), above 15% of RATED THERMAL POWER; adjust the linear power level, the CPC delta T power and CPC nuclear power ~~signals~~ to agree with the calorimetric calculation if the absolute difference is greater than 2%. During PHYSICS TESTS, these daily calibrations may be suspended provided these calibrations are performed upon reaching each major test power plateau and prior to proceeding to the next major test power plateau.

(3) - Above 15% of RATED THERMAL POWER, verify that the linear power subchannel gains of the excore detectors are consistent with the values used to establish the shape annealing matrix elements in the Core Protection Calculators.

(4) - Neutron detectors may be excluded from CHANNEL CALIBRATION.

(5) - After each fuel loading and prior to exceeding 70% of RATED THERMAL POWER, the incore detectors shall be used to determine the shape annealing matrix elements and the Core Protection Calculators shall use these elements.

(6) - This CHANNEL FUNCTIONAL TEST shall include the injection of simulated process signals into the channel as close to the sensors as practicable to verify OPERABILITY including alarm and/or trip functions.

(conservatively incorporated for measurement uncertainties)

(7) - Above 70% of RATED THERMAL POWER, verify that the total steady-state RCS flow rate as indicated by each CPC is less than or equal to the actual RCS total flow rate determined by either using the reactor coolant pump differential pressure instrumentation or by calorimetric calculations and if necessary, adjust the CPC addressable constant flow coefficients such that each CPC indicated flow is less than or equal to the actual flow rate. The flow measurement uncertainty may be included in the BERR1 term in the CPC and is equal to or greater than 4%.

(8) - Above 70% of RATED THERMAL POWER, verify that the total steady-state RCS flow rate as indicated by each CPC is less than or equal to the actual RCS total flow rate determined by either using the reactor coolant pump differentral pressure instrumentation and the ultrasonic flow meter adjusted pump curves or calorimetric calculations.

(9) - The monthly CHANNEL FUNCTIONAL TEST shall include verification that the correct values of addressable constants are installed in each OPERABLE CPC per Specification 2.2.2.

(10) - At least once per 18 months and following maintenance or adjustment of the reactor trip breakers, the CHANNEL FUNCTIONAL TEST shall include independent verification of the undervoltage and shunt trips.

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3/4.3.2 ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.2 The Engineered Safety Features Actuation System (ESFAS) instrumentation channels and bypasses shown in Table 3.3-3 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3-4 and with RESPONSE TIMES as shown in Table 3.3-5.

APPLICABILITY: As shown in Table 3.3-3.

ACTION:

a. With an ESFAS instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3-4, declare the channel inoperable and apply the applicable ACTION requirement of Table 3.3-3 until the channel is restored to OPERABLE status with the trip setpoint adjusted consistent with the Trip Setpoint value.

b. With an ESFAS instrumentation channel inoperable, take the ACTION shown in Table 3.3-3.

SURVEILLANCE REQUIREMENTS

4.3.2.1 Each ESFAS instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations for the MODES and at the frequencies shown in Table 4.3-2.

4.3.2.2 The logic for the bypasses shall be demonstrated OPERABLE during the at power CHANNEL FUNCTIONAL TEST of channels affected by bypass operation. The total bypass function shall be demonstrated OPERABLE at least once per 18 months during CHANNEL CALIBRATION testing of each channel affected by bypass operation.

4.3.2.3 The ENGINEERED SAFETY FEATURES RESPONSE TIME of each ESFAS function shall be demonstrated to be within the limit at least once per 18 months. Each test shall include at least one channel per function such that all channels are tested at least once every N times 18 months where N is the total number of redundant channels in a specific ESFAS function as shown in the "Total No. of Channels" Column of Table 3.3-3.

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TABLE 3.3-3

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

ESFA SYSTEM FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
I. SAFETY INJECTION (SIAS)					
A. Sensor/Trip Units					
1. Containment Pressure - High	4	2	3	1, 2, 3, 4	13*, 14*
2. Pressurizer Pressure - Low	4	2	3	1, 2, 3(a), 4	13*, 14*
B. ESFA System Logic					
1. Matrix Logic	6	1	3	1, 2, 3, 4	17
2. Initiation Logic	4(c)	2(d)	4	1, 2, 3, 4	12
3. Manual SIAS (Trip Buttons)	4(c)	2(d)	4	1, 2, 3, 4	12
C. Automatic Actuation Logic	2	1	2	1, 2, 3, 4	16
II. CONTAINMENT ISOLATION (CIAS)					
A. Sensor/Trip Units					
1. Containment Pressure - High	4	2	3	1, 2, 3	13*, 14*,
2. Pressurizer Pressure - Low	4	2	3	1, 2, 3(a)	13*, 14*
B. ESFA System Logic					
1. Matrix Logic	6	1	3	1, 2, 3	17
2. Initiation Logic	4(c)	2(d)	4	1, 2, 3, 4	12

CESSAR80 - NSSS-STS

3/4 3-16

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

ESFA SYSTEM FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
II. CONTAINMENT ISOLATION (Continued)					
3. Manual CIAS (Trip Buttons)	4(c)	2(d)	4	1, 2, 3, 4	12
~~4. Manual SIAS (Trip Buttons)~~	~~4(c)~~	~~2(d)~~	~~4~~	~~1, 2, 3, 4~~	~~12~~
C. Automatic Actuation Logic	2	1	2	1, 2, 3, 4	16
III. CONTAINMENT SPRAY (CSAS)					
A. Sensor/Trip Units					
Containment Pressure -- High - High	4	2	3	1, 2, 3	13*, 14*
B. ESFA System Logic					
1. Matrix Logic	6	1	3	1, 2, 3	17
2. Initiation Logic	4(c)	2(d)	4	1, 2, 3, 4	12
3. Manual CSAS (Trip Buttons)	4(c)	2(d)	4	1, 2, 3, 4	12
C. Automatic Actuation Logic	2	1	2	1, 2, 3, 4	16

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

ESFA SYSTEM FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
IV. MAIN STEAM LINE ISOLATION (MSIS)					
A. Sensor/Trip Units					
1. Steam Generator Pressure - Low	4/steam generator	2/steam generator	3/steam generator	1, 2, 3(b), 4	13*, 14*
2. Steam Generator Level - High	4/steam generator	2/steam generator	3/steam generator	1, 2, 3, 4	13*, 14*
3. Containment Pressure - High	4	2	3	1, 2, 3, 4	13*, 14*
B. ESFA System Logic					
1. Matrix Logic	6	1	3	1, 2, 3, 4	17
2. Initiation Logic	4(c)	2(d)	4	1, 2, 3, 4	12
3. Manual MSIS (Trip Buttons)	4(c)	2(d)	4	1, 2, 3, 4	12
C. Automatic Actuation Logic	2	1	2	1, 2, 3, 4	16

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TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

ESFA SYSTEM FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
V. RECIRCULATION (RAS)					
A. Sensor/Trip Units					
Refueling Water Storage Tank - Low	4	2	3	1, 2, 3	13*, 14*
B. ESFA System Logic					
1. Matrix Logic	6	1	3	1, 2, 3	17
2. Initiation Logic	4(c)	2(d)	4	1, 2, 3, 4	12
3. Manual RAS	4(c)	2(d)	4	1, 2, 3, 4	12
C. Automatic Actuation Logic	2	1	2	1, 2, 3, 4	16
VI. ~~AUXILIARY~~ EMERGENCY FEEDWATER (SG-1)(AFAS-1)					
A. Sensor/Trip Units					
1. Steam Generator #1 Level - Low	4	2	3	1, 2, 3	13*, 14*
2. Steam Generator Δ Pressure - SG2 > SG1	4	2	3	1, 2, 3	13*, 14*

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CESSAR 80-NSSS-575
PALO VERDE - UNIT 1

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

ESFA SYSTEM FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
VI. AUXILIARY FEEDWATER (SG-1)(AFAS-1) (Continued)					
B. ESFA System Logic					
1. Matrix Logic	6	1	3	1, 2, 3	17
2. Initiation Logic	4(c)	2(d)	4	1, 2, 3, 4	12
3. Manual AFAS	4(c)	2(d)	4	1, 2, 3, 4	15
C. Automatic Actuation Logic	2	1	2	1, 2, 3, 4	16
VII. ~~AUXILIARY~~ EMERGENCY FEEDWATER (SG-2)(AFAS-2)					
A. Sensor/Trip Units					
1. Steam Generator #2 Level - Low	4	2	3	1, 2, 3	13*, 14*
2. Steam Generator Δ Pressure - SG1 > SG2	4	2	3	1, 2, 3	13*, 14*
B. ESFA System Logic					
1. Matrix Logic	6	1	3	1, 2, 3	17
2. Initiation Logic	4(c)	2(d)	4	1, 2, 3, 4	12
3. Manual AFAS	4(c)	2(d)	4	1, 2, 3, 4	15
C. Automatic Actuation Logic	2	1	2	1, 2, 3, 4	16
VIII. LOSS OF POWER (LOV)					
A. 4.16 kV Emergency Bus Undervoltage (Loss of Voltage)	See Applicant's SAR ~~4/Bus~~	~~2/Bus~~	~~3/Bus~~	~~1, 2, 3~~	~~13*, 14*~~
B. 4.16 kV Emergency Bus Undervoltage (Degraded Voltage)	See Applicant's SAR ~~4/Bus~~	~~2/Bus~~	~~3/Bus~~	~~1, 2, 3~~	~~13*, 14*~~
IX. CONTROL ROOM ESSENTIAL FILTRATION	See Applicant's SAR ~~2~~	~~1~~	~~1~~	~~All Modes#~~	~~18~~

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TABLE NOTATIONS

(a) In MODES 3-6, the value may be decreased manually, to a minimum of 100 psia, as pressurizer pressure is reduced, provided the margin between the pressurizer pressure and this value is maintained at less than or equal to 400 psi; the setpoint shall be increased automatically as pressurizer pressure is increased until the trip setpoint is reached. Trip may be manually bypassed below 400 psia; bypass shall be automatically removed whenever pressurizer pressure is greater than or equal to 500 psia.

(b) In MODES 3-6, the value may be decreased manually as steam generator pressure is reduced, provided the margin between the steam generator pressure and this value is maintained at less than or equal to 200 psi; the setpoint shall be increased automatically as steam generator pressure is increased until the trip setpoint is reached.

(c) Four channels provided, arranged in a selective two-out-of-four configuration (i.e., one-out-of-two taken twice).

(d) The proper two-out-of-four combination.

* The provisions of Specification 3.0.4 are not applicable.

~~# After the initial criticality of Unit 2 or Unit 3.~~

ACTION STATEMENTS

ACTION 12 - With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

ACTION 13 - With the number of channels OPERABLE one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may continue provided the inoperable channel is placed in the bypassed or tripped condition within 1 hour. If the inoperable channel is bypassed, the desirability of maintaining this channel in the bypassed condition shall be reviewed in accordance with Specification 6.5.1.6.~~h~~ of Administrative Controls. The channel shall be returned to OPERABLE status no later than during the next COLD SHUTDOWN.

With a channel process measurement circuit that affects multiple functional units inoperable or in test, bypass or trip all associated functional units as listed below.

Process Measurement Circuit	
1. Steam Generator Pressure - Low	Steam Generator Pressure - Low (RPS) Steam Generator Level 1-Low (ESF) Steam Generator Level 2-Low (ESF)
2. Steam Generator Level (Wide Range)	Steam Generator Level - Low (RPS) Steam Generator Level 1-Low (ESF) Steam Generator Level 2-Low (ESF)

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TABLE 3.3-3 (Continued)

ACTION STATEMENTS (Continued)

ACTION 14 - With the number of channels OPERABLE one less than the Minimum Channels OPERABLE, STARTUP and/or POWER OPERATION may continue provided the following conditions are satisfied:

a. Verify that one of the inoperable channels has been bypassed and place the other inoperable channel in the tripped condition within 1 hour.

b. All functional units affected by the bypassed/tripped channel shall also be placed in the bypassed/tripped condition as listed below:

Process Measurement Circuit	Functional Unit Bypassed/Tripped
1. Steam Generator Pressure - Low	Steam Generator Pressure - Low (RPS) Steam Generator Level 1 - Low (ESF) Steam Generator Level 2 - Low (ESF)
2. Steam Generator Level - Low (Wide Range)	Steam Generator Level - Low (RPS) Steam Generator Level 1 - Low (ESF) Steam Generator Level 2 - Low (ESF)

STARTUP and/or POWER OPERATION may continue until the performance of the next required CHANNEL FUNCTIONAL TEST. Subsequent STARTUP and/or POWER OPERATION may continue if one channel is restored to OPERABLE status and the provisions of ACTION 14 are satisfied.

ACTION 15 - With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

ACTION 16 - With the number of OPERABLE channels one less than the Total Number of Channels, be in at least HOT STANDBY within 6 hours and in at least HOT SHUTDOWN within the following 6 hours; however, one channel may be bypassed for up to 1 hour for surveillance testing provided the other channel is OPERABLE.

ACTION 17 - With the number of OPERABLE channels one less than the Minimum Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

~~ACTION 18 - With the number of OPERABLE channels one less than the Minimum Number of Channels, operation may continue for up to 6 hours. After 6 hours operation may continue provided at least 1 train of essential filtration is in operation, otherwise, be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.~~

CESSAR80-NSSS-STS ~~PALO VERDE - UNIT 1~~ 3/4 3-22

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TABLE 3.3-4

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP VALUES*

ESFA SYSTEM FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
I. SAFETY INJECTION (SIAS)		
A. Sensor/Trip Units		
1. Containment Pressure - High	≤ [redacted] psig	≤ [redacted] psig
2. Pressurizer Pressure - Low	≥ [redacted] psia(1)	≥ [redacted] psia(1)
B. ESFA System Logic	Not Applicable	Not Applicable
C. Actuation Systems	Not Applicable	Not Applicable
II. CONTAINMENT ISOLATION (CIAS)		
A. Sensor/Trip Units		
1. Containment Pressure - High	≤ [redacted] psig	≤ [redacted] psig
2. Pressurizer Pressure - Low	≥ [redacted] psia(1)	≥ [redacted] psia(1)
B. ESFA System Logic	Not Applicable	Not Applicable
C. Actuation Systems	Not Applicable	Not Applicable
III. CONTAINMENT SPRAY (CSAS)		
A. Sensor/Trip Units		
Containment Pressure High - High	≤ [redacted] psig	≤ [redacted] psig
B. ESFA System Logic	Not Applicable	Not Applicable
C. Actuation Systems	Not Applicable	Not Applicable
IV. MAIN STEAM LINE ISOLATION (MSIS)		
A. Sensor/Trip Units		
1. Steam Generator Pressure - Low	≥ [redacted] psia(3)	≥ [redacted] psia(3)
2. Steam Generator Level - High	≤ [redacted] NR(2)	≤ [redacted] NR(2)
3. Containment Pressure - High	≤ [redacted] psig	≤ [redacted] psig
B. ESFA System Logic	Not Applicable	Not Applicable
C. Actuation Systems	Not Applicable	Not Applicable

CESSAR80-NSSS-STS

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP VALUES*

ESFA SYSTEM FUNCTIONAL UNIT	TRIP VALUES	ALLOWABLE VALUES
V. RECIRCULATION (RAS)		
A. Sensor/Trip Units		
Refueling Water Storage Tank - Low	≥ [redacted]% of Span	[redacted] ≥ % of Span ≥ [redacted]
B. ESFA System Logic	Not Applicable	Not Applicable
C. Actuation System	Not Applicable	Not Applicable
VI. ~~AUXILIARY~~ EMERGENCY FEEDWATER (SG-1)(EFAS-1)		
A. Sensor/Trip Units		
1. Steam Generator #1 Level - Low	≥ [redacted]% WR(4)	≥ [redacted]% WR(4)
2. Steam Generator Δ Pressure - SG2 > SG1	≤ [redacted] psid	≤ [redacted] psid
B. ESFA System Logic	Not Applicable	Not Applicable
C. Actuation Systems	Not Applicable	Not Applicable
VII. ~~AUXILIARY~~ FEEDWATER (SG-2)(EFAS-2)		
A. Sensor/Trip Units		
1. Steam Generator #2 Level - Low	≥ [redacted]% WR(4)	≥ [redacted]% WR(4)
2. Steam Generator Δ Pressure - SG1 > SG2	≤ [redacted] psid	≤ [redacted] psid
B. ESFA System Logic	Not Applicable	Not Applicable
C. Actuation Systems	Not Applicable	Not Applicable
VIII. LOSS OF POWER**		
A. 4.16 kV Emergency Bus Undervoltage (Loss of Voltage)	≥ [redacted] volts	≥ [redacted] volts
B. 4.16 kV Emergency Bus Undervoltage (Degraded Voltage)	[redacted] to [redacted] volts with a 35-second maximum time delay	[redacted] to [redacted] volts with a 35-second maximum time delay
IX. CONTROL ROOM ESSENTIAL FILTRATION**	≥ [redacted] μCi/cc	≤ [redacted] μCi/cc

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TABLE 3.3-4 (Continued)

TABLE NOTATIONS

(1) In MODES 3-6, value may be decreased manually, to a minimum of 100 psia, as pressurizer pressure is reduced, provided the margin between the pressurizer pressure and this value is maintained at less than or equal to 400 psi; the setpoint shall be increased automatically as pressurizer pressure is increased until the trip setpoint is reached. Trip may be manually bypassed below 400 psia; bypass shall be automatically removed whenever pressurizer pressure is greater than or equal to 500 psia.

(2) % of the distance between steam generator upper and lower level narrow range instrument nozzles.

(3) In MODES 3-6, value may be decreased manually as steam generator pressure is reduced, provided the margin between the steam generator pressure and this value is maintained at less than or equal to 200 psi; the setpoint shall be increased automatically as steam generator pressure is increased until the trip setpoint is reached.

(4) % of the distance between steam generator upper and lower level wide range instrument nozzles.

* See Applicant's SAR

** Balance of Plant (BOP)

TABLE 3.3-5

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ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
1. Manual	
a. SIAS	
Safety Injection (ECCS)	Not Applicable
Containment Isolation	Not Applicable (BOP)
Containment Purge Valve Isolation	Not Applicable (BOP)
b. CSAS	
Containment Spray	Not Applicable (BOP)
c. CIAS	
Containment Isolation	Not Applicable (BOP)
~~Containment Purge Valve Isolation~~	~~Not Applicable (BOP)~~
d. MSIS	
Main Steam Isolation	Not Applicable
e. RAS	
Containment Sump Recirculation	Not Applicable (BOP)
f. EFAS	
~~Auxiliary~~ Emergency Feedwater Pumps	Not Applicable (BOP)
g. CCAS	
Containment Cooling	Not Applicable (BOP)

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TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
2. Pressurizer Pressure - Low	
a. Safety Injection (HPSI)	≤ [redacted]*/[redacted]**
b. Safety Injection (LPSI)	≤ [redacted]*/[redacted]
c. Containment Isolation	≤ */ **
~~1. CIAS actuated mini-purge valves~~	~~≤ [redacted]**~~
~~2. Other CIAS actuated valves~~	~~≤ [redacted]*/[redacted]**~~
3. Containment Pressure - High	
a. Safety Injection (HPSI)	≤ [redacted]*/[redacted]**
b. Safety Injection (LPSI)	≤ [redacted]*/[redacted]**
c. Containment Isolation	≤ */ **
~~1. CIAS actuated minipurge valves~~	~~≤ [redacted]**~~
~~2. Other CIAS actuated valves~~	~~≤ [redacted]*/[redacted]**~~
d. Main Steam Isolation	≤ */ **
~~1. MSIS actuated MSIV's~~	~~≤ [redacted]~~
~~2. MSIS actuated MFIV's~~	~~≤ [redacted]**~~
e. Containment Spray Pump	≤ [redacted]*/[redacted]**
4. Containment Pressure - High-High	
a. Containment Spray	≤ [redacted]*/[redacted]**
5. Steam Generator Pressure - Low	
a. Main Steam Isolation	≤ */ **
~~1. MSIS actuated MSIV's~~	~~≤ [redacted]**~~
~~2. MSIS actuated MFIV's~~	~~≤ [redacted]**~~
6. Refueling Water Tank - Low	
a. Containment Sump Recirculation	≤ [redacted]*/[redacted]**
7. Steam Generator Level - Low	
a. ~~Auxiliary~~ Emergency Feedwater ~~(Motor Drive)~~	≤ [redacted]*/[redacted]**
~~b. Auxiliary Feedwater (turbine drive)~~	~~≤ [redacted]*/[redacted]**~~

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TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES[1]

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
8. Steam Generator Level - High	
a. Main Steam Isolation	≤ */ **
~~1. MSIS actuated MSIV's~~	~~≤ [redacted]*/[redacted]**~~
~~2. MSIS actuated MFIV's~~	~~≤ [redacted]*/[redacted]**~~
9. Steam Generator ΔP-High-Coincident With Steam Generator Level Low	
a. Emergency Feedwater Isolation from the Ruptured Steam Generator	≤ [redacted]*/[redacted]**
10. Control Room Essential Filtration Actuation[2]	≤ [redacted]*/≤ [redacted]** [redacted]
11. 4.16 kV Emergency Bus Undervoltage[2] (Degraded Voltage)	[redacted]
Loss of Power 90% system voltage	≤ [redacted]
12. 4.16 kV Emergency Bus Undervoltage (loss of Voltage)[2]	[redacted]
Loss of Power	≤ [redacted]

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TABLE NOTATIONS

*Diesel generator starting and sequence loading delays included. Response time limit includes movement of valves and attainment of pump or blower discharge pressure.

**Diesel generator starting delays not included. Offsite power available. Response time limit includes movement of valves and attainment of pump or blower discharge pressure.

~~#MFIV valves tested at simulated operating conditions; valves tested at static flow conditions to ≤ 8.6/8.6 seconds.~~

1 See Applicant's SAR for Response Time(s).
2 Balance of Plant (BOP)

~~PALO VERDE - UNIT 1~~ CESSAR 80-NSSS-STS

3/4 3-~~X~~ 22

TABLE 4.3-2

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

ESFA SYSTEM FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
I. SAFETY INJECTION (SIAS)				
A. Sensor/Trip Units				
1. Containment Pressure - High	S	R	M	1, 2, 3, 4
2. Pressurizer Pressure - Low	S	R	M	1, 2, 3, 4
B. ESFA System Logic				
1. Matrix Logic	NA	NA	M	1, 2, 3, 4
2. Initiation Logic	NA	NA	M	1, 2, 3, 4
3. Manual SIAS	NA	NA	M	1, 2, 3, 4
C. Automatic Actuation Logic	NA	NA	M(1) (2) (3)	1, 2, 3, 4
II. CONTAINMENT ISOLATION (CIAS)				
A. Sensor/Trip Units				
1. Containment Pressure - High	S	R	M	1, 2, 3
2. Pressurizer Pressure - Low	S	R	M	1, 2, 3
B. ESFA System Logic				
1. Matrix Logic	NA	NA	M	1, 2, 3, 4
2. Initiation Logic	NA	NA	M	1, 2, 3, 4
3. Manual CIAS	NA	NA	M	1, 2, 3, 4
~~4. Manual SIAS~~	~~NA~~	~~NA~~	~~M~~	~~1, 2, 3, 4~~

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TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

ESFA SYSTEM FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
II. CONTAINMENT ISOLATION (Continued)				
C. Automatic Actuation Logic	NA	NA	M(1) (2) (3)	1, 2, 3, 4
III. CONTAINMENT SPRAY (CSAS)				
A. Sensor/Trip Units				
1. Containment Pressure -- High - High	S	R	M	1, 2, 3
B. ESFA System Logic				
1. Matrix Logic	NA	NA	M	1, 2, 3, 4
2. Initiation Logic	NA	NA	M	1, 2, 3, 4
3. Manual CSAS	NA	NA	M	1, 2, 3, 4
C. Automatic Actuation Logic	NA	NA	M(1) (2) (3)	1, 2, 3, 4

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

ESFA SYSTEM FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
IV. MAIN STEAM LINE ISOLATION (MSIS)				
A. Sensor/Trip Units				
1. Steam Generator Pressure - Low	S	R	M	1, 2, 3, 4
2. Steam Generator Level - High	S	R	M	1, 2, 3, 4
3. Containment Pressure - High	S	R	M	1, 2, 3, 4
B. ESFA System Logic				
1. Matrix Logic	NA	NA	M	1, 2, 3, 4
2. Initiation Logic	NA	NA	M	1, 2, 3, 4
3. Manual MSIS	NA	NA	M	1, 2, 3, 4
C. Automatic Acutation Logic	NA	NA	M(1) (2) (3)	1, 2, 3, 4

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TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

ESFA SYSTEM FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
V. RECIRCULATION (RAS)				
A. Sensor/Trip Units				
Refueling Water Storage Tank - Low	S	R	M	1, 2, 3
B. ESFA System Logic				
1. Matrix Logic	NA	NA	M	1, 2, 3, 4
2. Initiation Logic	NA	NA	M	1, 2, 3, 4
3. Manual RAS	NA	NA	M	1, 2, 3, 4
C. Automatic Acutation Logic	NA	NA	M(1) (2) (3)	1, 2, 3, 4
VI. ~~AUXILIARY~~ EMERGENCY FEEDWATER (SG-1)(~~A~~EFAS-1)				
A. Sensor/Trip Units				
1. Steam Generator #1 Level - Low	S	R	M	1, 2, 3
2. Steam Generator Δ Pressure SG2 > SG1	S	R	M	1, 2, 3

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EMERGENCY

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

ESFA SYSTEM FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
VI. ~~AUXILIARY~~ FEEDWATER (SG-1)(~~A~~EFAS-1) (Continued)				
B. ESFA System Logic				
1. Matrix Logic	NA	NA	M	1, 2, 3, 4
2. Initiation Logic	NA	NA	M	1, 2, 3, 4
3. Manual AFAS	NA	NA	M	1, 2, 3, 4
C. Automatic Actuation Logic	NA	NA	M(1) (2) (3)	1, 2, 3, 4
VII. ~~AUXILIARY~~ FEEDWATER (SG-2)(~~A~~EFAS-2)				
A. Sensor/Trip Units				
1. Steam Generator #2 Level - Low	S	R	M	1, 2, 3
2. Steam Generator Δ Pressure SG1 > SG2	S	R	M	1, 2, 3
B. ESFA System Logic				
1. Matrix Logic	NA	NA	M	1, 2, 3, 4
2. Initiation Logic	NA	NA	M	1, 2, 3, 4
3. Manual AFAS	NA	NA	M	1, 2, 3, 4
C. Automatic Actuation Logic	NA	NA	M(1) (2) (3)	1, 2, 3, 4
VIII. LOSS OF POWER (LOV)				
A. ~~4.16 KV Emergency Bus Undervoltage (Loss of Voltage)~~ (Loss of Voltage)	~~S~~ See Applicant's SAR	~~R~~		~~1, 2, 3, 4~~ (BOP)
B. ~~4.16 KV Emergency Bus Undervoltage~~ (Degraded Voltage)	~~S~~ See Applicant's SAR	~~R~~		~~1, 2, 3, 4~~ (BOP)

TABLE 4.3-2 (Continued)

TABLE NOTATION

(1) Each train or logic channel shall be tested at least every 62 days on a STAGGERED TEST BASIS.

(2) Testing of automatic actuation logic shall include energization/deenergization of each initiation relay and verification of proper operation of each initiation relay.

(3) A subgroup relay test shall be performed which shall include the energization/deenergization of each subgroup relay and verification of the OPERABILITY of each subgroup relay. ~~Relays listed below are exempt from testing during POWER OPERATION but shall be tested at least once per 18 months during REFUELING and during each COLD SHUTDOWN condition unless tested within the previous 62 days.~~

~~ACTUATION DEVICES THAT CANNOT BE TESTED AT POWER~~

TRAIN A		TRAIN B	
ESF FUNCTION	ACTUATION DEVICE	ESF FUNCTION	ACTUATION DEVICE
SIAS A	K108	SIAS B	K208
SIAS A	K409	SIAS B	K409
CIAS A	K202	CIAS B	K204
CIAS A	K204	CIAS B	K205
CIAS A	K205	CSAS B	K304
CSAS A	K304	MSIS B	K305
MSIS A	K305	MSIS B	K404
MSIS A	K404	AFAS 1B	K113
AFAS 1A	K211	AFAS 1B	K211
AFAS 2A	K112	AFAS 2B	K112

In the case of the following relays which are tested during power operation, ~~one or more pieces of equipment cannot be actuated, but can be racked out,~~ bypassed or etc., which will not preclude the relay from being tested but will not actuate the locked out equipment associated with the relay:

SIAS A	K401	SIAS B	K301
SIAS A	K410	SIAS B	K308
SIAS A	K412	CIAS B	K203
CIAS A	K203	CIAS B	K210
CIAS A	K210	RAS B	K104
RAS A	K104	RAS B	K312
RAS A	K312	RAS B	K405
RAS A	K405		
AFAS 1A	K113		

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3/4.3.1 Radiation Monitoring Instrumentation

See Applicant's SAR

3/4.3.3 MONITORING INSTRUMENTATION

RADIATION MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.1 The radiation monitoring instrumentation channels shown in Table 3.3-6 shall be OPERABLE with their alarm/trip setpoints within the specified limits.

APPLICABILITY: As shown in Table 3.3-6.

ACTION:

a. With a radiation monitoring channel alarm/trip setpoint exceeding the value shown in Table 3.3-6, adjust the setpoint to within the limit within 4 hours or declare the channel inoperable.

b. With the number of channels OPERABLE one less than the Minimum Channels OPERABLE requirement, take the ACTION shown in Table 3.3-6.

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

SURVEILLANCE REQUIREMENTS

4.3.3.1 Each radiation monitoring instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION, and CHANNEL FUNCTIONAL TEST operations for the MODES and at the frequencies shown in Table 4.3-3.

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LIMITING CONDITION FOR OPERATION

3.3.3.2 The incore detection system shall be OPERABLE with:

a. At least 75% of all incore detector locations, and

b. A minimum of two quadrant symmetric incore detector locations per core quadrant.

An OPERABLE incore detector location shall consist of a fuel assembly containing a fixed detector string with a minimum of four OPERABLE rhodium detectors or an OPERABLE movable incore detector capable of mapping the location.

APPLICABILITY: When the incore detection system is used for monitoring:

a. AZIMUTHAL POWER TILT,

b. Radial Peaking Factors,

c. Local Power Density,

d. DNB Margin.

ACTION:

a. With the incore detection system inoperable, do not use the system for the above applicable monitoring or calibration functions.

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

SURVEILLANCE REQUIREMENTS

4.3.3.2 The incore detection system shall be demonstrated OPERABLE:

a. By performance of a CHANNEL CHECK within 24 hours prior to its use if the system has just been returned to OPERABLE status or if 7 days or more have elapsed since last use and at least once per 7 days thereafter when required for monitoring the AZIMUTHAL POWER TILT, radial peaking factors, local power density or DNB margin:

b. At least once per 18 months by performance of a CHANNEL CALIBRATION operation which exempts the neutron detectors but includes all electronic components. The fixed incore neutron detectors shall be calibrated prior to installation in the reactor core.

LIMITING CONDITION FOR OPERATION AND SURVEILLANCE REQUIREMENTS

INSERT

~~3.3.3.3 The seismic monitoring instrumentation shown in Table 3.3-7 shall be~~ OPERABLE.

APPLICABILITY: At all times.

ACTION:

a. With one or more seismic monitoring instruments inoperable 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

4.3.3.3.1 Each of the above 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 4.3-4.

4.3.3.3.2 Each of the above seismic monitoring instruments actuated during a seismic event (greater than or equal to 0.02g) shall have a CHANNEL CALIBRATION performed within 5 days. Data shall be retrieved from actuated instruments and analyzed to determine the magnitude of the vibratory ground motion. A Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 10 days describing the magnitude, frequency spectrum, and resultant effect upon facility features important to safety.

Insert

[redacted]

3/4.3.3.3 Seismic Monitoring Instrumentation

Table 3.3-7, Seismic Monitoring Instrumentation

Table 4.3-5, Seismic Monitoring Instrumentation Surveillance Requirements

/4.3.3.4 Meteorological Monitoring Instrumentation

Table 3.3-8, Meteorological Monitoring Instrumentation

Table 4.3-5, Meteorological Monitoring Instrumentation Surveillance Requirements

See Applicant's SAR

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REMOTE SHUTDOWN SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.5 The remote shutdown system ~~disconnect switches, power,~~ controls and monitoring instrumentation channels shown in Table 3.3-9 shall be OPERABLE.

APPLICABILITY: MODES 1 ~~and~~ 2, and 3

ACTION:

a. With the number of OPERABLE remote shutdown monitoring channels less than required by Table 3.3-9, restore the inoperable channel(s) to OPERABLE status within 7 days, or be in HOT [redacted] SHUTDOWN within the next 12 hours.

b. With one or more remote required by Table 3.3-9 shutdown system instrumentation [redacted] control circuits inoperable, restore the inoperable ~~switch(s)/~~ circuit(s) to OPERABLE status ~~or issue procedure changes per Specification 6.8.3 that identifies alternate disconnect methods or power or control circuits for remote shutdown~~ within 7 days, or be in HOT [redacted] SHUTDOWN within the next 12 hours.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.5 The ~~R~~emote ~~S~~hutdown ~~S~~ystem shall be demonstrated ~~OPERABLE~~:

a. By performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3-6 for each remote shutdown monitoring instrumentation channel.

b. By operation of each remote shutdown system instrumentation ~~disconnect switch and~~ [redacted] control circuit including the actuated components at least once per 18 months.

TABLE 3.3-9

REMOTE SHUTDOWN INSTRUMENTATION~~, DISCONNECT SWITCHES AND CONTROL CIRCUITS~~

INSTRUMENT~~ATION~~	READOUT LOCATION	MINIMUM CHANNELS OPERABLE
1. Log Neutron Power Level	Remote Shutdown Panel	(2)
2. Reactor Coolant Hot Leg Temperature	Remote Shutdown Panel	(1/loop)
3. Reactor Coolant Cold Leg Temperature	Remote Shutdown Panel	(1/loop)
4. Pressurizer Pressure	Remote Shutdown Panel	(1)
5. Pressurizer Level	Remote Shutdown Panel	(2)
6. Steam Generator Pressure	Remote Shutdown Panel	(2/steam generator)
7. Steam Generator Level	Remote Shutdown Panel	(2/steam generator)
8. Refueling Water Tank Level	Remote Shutdown Panel	(2)
9. Charging Line Pressure	Remote Shutdown Panel	(1)
10. Charging Line Flow	Remote Shutdown Panel	(1)
11. Shutdown Cooling Heat Exchanger Temperatures	Remote Shutdown Panel	(2)
12. Shutdown Cooling Flow	Remote Shutdown Panel	(2)
13. Auxiliary Feedwater Flow Rate	Remote Shutdown Panel	(2/steam generator)

CONTROL CIRCUIT	SWITCH LOCATION
1. Pressurizer Heater	
2. Auxiliary Feedwater (BOP)	
3. Diesel Generator (BOP)	
4. Pressurizer Auxiliary Spray	
5. Essential chilled Water	
6. Safety Injection Tank	
7.	
8.	

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TABLE 4.3-6

REMOTE SHUTDOWN INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. Log Neutron Power Level	M	R
2. Reactor Coolant Hot Leg Temperature (2)	M	R
3. Reactor Coolant Cold Leg Temperature (2)	M	R
4. Pressurizer Pressure	M	R
5. Pressurizer Level	M	R
6. Steam Generator Pressure	M	R
7. Steam Generator Level	M	R
8. Refueling Water Tank Level	M	R
9. Charging Line Pressure	M	R
10. Charging Line Flow	M	R
11. Shutdown Cooling Heat Exchanger Temperatures	M	R
12. Shutdown Cooling Flow	M	R
13. ~~Auxiliary~~ Emergency Feedwater Flow Rate	M	R

(BOP)

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POST-ACCIDENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.6 The post-accident monitoring instrumentation channels shown in Table 3.3-10 shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With one or more accident monitoring instrumentation channels inoperable, take the action shown in Table 3.3-10.

b. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.6 Each post-accident monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3-7.

TABLE 3.3-10

POST-MONITORING INSTRUMENTATION

INSTRUMENT	REQUIRED NUMBER OF CHANNELS	MINIMUM CHANNELS OPERABLE	ACTION
1. Containment Pressure	2	1	29,30
2. Reactor Coolant Outlet Temperature - T_{hot} (Wide Range)	2	1/loop	29,30
3. Reactor Coolant Inlet Temperature - T_{cold} (Wide Range)	2	1/loop	
4. Pressurizer Pressure - Wide Range	2	1	29,30
5. Pressurizer Water Level	2	1	29,30
6. Steam Generator Pressure	2/steam generator	1/steam generator	29,30
7. Steam Generator Water Level - Wide Range	2/steam generator	1/steam generator	29,30
8. Refueling Water Storage Tank Water Level	2	1	29,30
9. Auxiliary Feedwater Flow Rate	2	1	29,30
10. Reactor Cooling System Subcooling Margin Monitor	2	1	29,30
11. Pressurizer Safety Valve Position Indicator	1/valve	1/valve	29,30
12. Containment Water Level (Narrow Range)	2	1	29,30
13. Containment Water Level (Wide Range)	2	1	29,30
14. Core Exit Thermocouples	4/core quadrant	2/core quadrant	29,30
15. Reactor Vessel Water Level	2*	1*	29,30
16. Neutron Flux Monitor (Power Range)	2	1	29,30
17. Unit Vent - High Range Noble Gas Monitor	NA	1	31 (BOP)
18. Containment Atmosphere - High Range Radiation Monitor	NA	1	31 (BOP)

*A channel is eight sensors in a probe. A channel is OPERABLE if four or more sensors, two or more in the upper four and two or more in the lower four, are OPERABLE.

TABLE ~~4.3-7~~ 3.3-10
ACTION STATEMENTS

ACTION 29 - With the number of OPERABLE Channels one less than the Required Number of Channels in Table 3.3-10, either restore the Inoperable Channel(s) to OPERABLE status within 7 days, or be in HOT SHUTDOWN within the next 12 hours.

ACTION 30 - With the number of OPERABLE Channels one less than the Minimum Channels OPERABLE in Table 3.3-10, either restore the Inoperable Channel(s) to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN within the next 12 hours.

~~ACTION 31 - With the number of OPERABLE Channels one less than the Required Number of Channels, either restore the system to OPERABLE status within 7 days if repairs are feasible without shutting down or prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.~~

ACTION ~~32~~ 31 - With the number of OPERABLE Channels one less than the Minimum Channels OPERABLE in Table 3.3-10, either restore the inoperable channel(s) to OPERABLE status within 48 hours if repairs are feasible without shutting down or:

1. Initiate an alternate method of monitoring the ~~reactor~~ appropriate parameter(s) ~~vessel inventory~~;

2. Prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status; and

3. Restore the system to OPERABLE status at the next scheduled refueling.

TABLE 4.3-7

POST-ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION	
1. Containment Pressure	M	R	
2. Reactor Coolant Outlet Temperature - T_{hot} (Wide Range)	M	R	
3. Reactor Coolant Inlet Temperature - T_{cold} (Wide Range)	M	R	
4. Pressurizer Pressure - Wide Range	M	R	
5. Pressurizer Water Level	M	R	
6. Steam Generator Pressure	M	R	
7. Steam Generator Water Level - Wide Range	M	R	
8. Refueling Water Storage Tank Water Level	M	R	
9. Auxiliary Feedwater Flow Rate	M	R	
10. Reactor Coolant System Subcooling Margin Monitor	M	R	
11. Pressurizer Safety Valve Position Indicator	M	R	
12. Containment Water Level (Narrow Range)	M	R	
13. Containment Water Level (Wide Range)	M	R	
14. Core Exit Thermocouples	M	R	
15. Reactor Vessel Water Level	M	R	
16. Neutron Flux Monitor (Power Range)	M	R	
17. Unit Vent-High Range Noble Gas Monitor	M	R	(BOP)
18. Containment Atmosphere - High Range Radiation Monitor	M	R	(BOP)

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CHLORINE DETECTION,

INSTRUMENTATION , LOOSE-PART DETECTION, AND RADIOACTIVE GASEOUS EFFLUENT MONITORING

~~FIRE DETECTION~~ INSTRUMENTATION

LIMITING CONDITION FOR OPERATION AND SURVEILLANCE REQUIREMENTS

~~3.3.3.7 As a minimum, the fire detection instrumentation for each fire~~ detection zone shown in Table 3.3-11 shall be OPERABLE.

APPLICABILITY: Whenever equipment protected by the fire detection instrument is required to be OPERABLE.

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ACTION:

a. With any, but not more than one-half the total in any fire zone Function X fire detection instrument shown in Table 3.3-11 inoperable, restore the inoperable instrument(s) to OPERABLE status within 14 days or within the next 1 hour establish a fire watch patrol to inspect the zone(s) with the inoperable instrument(s) at least once per hour, unless the instrument(s) is located inside the containment, then inspect that containment zone at least once per 8 hours or monitor the containment air temperature at least once per hour at the locations listed in Specification 4.6.1.5.

b. With more than one-half of the Function X fire detection instruments in any fire zone shown in Table 3.3-11 inoperable, or with any Function Y fire detection instruments shown in Table 3.3-11 inoperable, or with any two or more adjacent fire detection instruments shown in Table 3.3-11 inoperable, within 1 hour establish a fire watch patrol to inspect the zone(s) with the inoperable instrument(s) at least once per hour, unless the instrument(s) is located inside the containment, then inspect that containment zone at least once per 8 hours or monitor the containment air temperature at least once per hour at the locations listed in Specification 4.6.1.5.

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

INSERT 1

SURVEILLANCE REQUIREMENTS

4.3.3.7.1 Each of the above required fire detection instruments which are accessible during plant operation shall be demonstrated OPERABLE at least once per 6 months by performance of a CHANNEL FUNCTIONAL TEST. Fire detectors which are not accessible during plant operation shall be demonstrated OPERABLE by the performance of a CHANNEL FUNCTIONAL TEST during each COLD SHUTDOWN exceeding 24 hours unless performed in the previous 6 months.

4.3.3.7.2 The NFPA Standard 72D supervised circuits supervision associated with the detector alarms of each of the above required fire detection ~~instruments shall be demonstrated OPERABLE at least once per 6 months.~~

Insert

3/4.3.3.7 Chlorine Detection Systems

3/4.3.3.8 Fire Detection Instrumentation

Table 3.3-11, Fire Detection Instruments

3/4.3.3.9 Loose-Part Detection Instrumentation

Table 3.3-12, Loose Parts Sensor Locations

(Instrumentation)

3/4.3.3.10 Radioactive Gaseous Effluent Monitoring

Table 3.3-13, Radioactive Gaseous Effluents Monitoring Instrumentation

Table 4.3-8, Radioactive Gaseous Effluent Monitoring Instrumentation Surveillance Requirements

See [illegible]licant's SAR

TURBINE OVERSPEED PROTECTION

LIMITING CONDITION FOR OPERATION AND SURVEILLANCE REQUIREMENTS

3/4.3.4 Turbine Overspeed Protection

See Applicant's SAR.

3/4.4 REACTOR COOLANT SYSTEM

3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

STARTUP AND POWER OPERATION

LIMITING CONDITION FOR OPERATION

3.4.1.1 Both reactor coolant loops and both reactor coolant pumps in each loop shall be in operation.

APPLICABILITY: MODES 1 and 2.*

ACTION:

With less than the above required reactor coolant pumps in operation, be in at least HOT STANDBY within 1 hour.

SURVEILLANCE REQUIREMENTS

4.4.1.1 The above required reactor coolant loops shall be verified to be in operation and circulating reactor coolant at least once per 12 hours.

*See Special Test Exception 3.10.3.

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LIMITING CONDITION FOR OPERATION

3.4.1.2 The reactor coolant loops listed below shall be OPERABLE and at least one of these reactor coolant loops shall be in operation.*

a. Reactor Coolant Loop 1 and its associated steam generator and at least one associated reactor coolant pump.

b. Reactor Coolant Loop 2 and its associated steam generator and at least one associated reactor coolant pump.

APPLICABILITY: MODE 3.

ACTION:

a. 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.

b. 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.

SURVEILLANCE REQUIREMENTS

4.4.1.2.1 At least the above required reactor coolant pumps, if not in operation, shall be determined to be OPERABLE once per 7 days by verifying correct breaker alignments and indicated power availability.

4.4.1.2.2 At least one reactor coolant loop shall be verified to be in operation and circulating reactor coolant at least once per 12 hours.

4.4.1.2.3 The required steam generator(s) shall be determined OPERABLE verifying the secondary side water level to be ≥ 25% indicated wide range level at least once per 12 hours.

*All reactor coolant pumps may be deenergized for up to 1 hour provided (1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

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LIMITING CONDITION FOR OPERATION

3.4.1.3 At least two of the loop(s)/train(s) listed below shall be OPERABLE and at least one reactor coolant and/or shutdown cooling loops shall be in operation.*

a. Reactor Coolant Loop 1 and its associated steam generator and at least one associated reactor coolant pump,**

b. Reactor Coolant Loop 2 and its associated steam generator and at least one associated reactor coolant pump,**

c. Shutdown Cooling Train ~~A~~, #1 X

d. Shutdown Cooling Train ~~B~~. #2 X

APPLICABILITY: MODE 4.

ACTION:

a. With less than the above required reactor coolant and/or shutdown cooling loops OPERABLE, immediately initiate corrective action to return the required loops to OPERABLE status as soon as possible; if the remaining OPERABLE loop is a shutdown cooling loop, be in COLD SHUTDOWN within 24 hours.

b. With no reactor coolant or shutdown cooling 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 coolant loop to operation.

*All reactor coolant pumps and shutdown cooling pumps may be deenergized for up to 1 hour provided (1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

A reactor coolant pump shall not be started with one or more * of the Reactor Coolant System cold leg temperatures less than or equal to ~~[illegible]~~°F during cooldown, or ~~[illegible]~~°F during heatup, unless the secondary water temperature (saturation temperature corresponding to steam generator pressure) of each steam generator is less than 100°F above each of the Reactor Coolant System cold leg temperatures. X X

*** See Applicant's SAR

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SURVEILLANCE REQUIREMENTS

4.4.1.3.1 The required reactor coolant pump(s), if not in operation, shall be determined to be OPERABLE once per 7 days by verifying correct breaker alignments and indicated power availability.

4.4.1.3.2 The required steam generator(s) shall be determined OPERABLE by verifying the secondary side water level to be ≥ 25% indicated wide range level at least once per 12 hours.

4.4.1.3.3 At least one reactor coolant or shutdown cooling loop shall be verified to be in operation and circulating reactor coolant at a flow rate greater than or equal to 4000 gpm at least once per 12 hours.

REACTOR COOLANT SYSTEM

COLD SHUTDOWN - LOOPS FILLED

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LIMITING CONDITION FOR OPERATION

3.4.1.4.1 At least one shutdown cooling loop shall be OPERABLE and in operation*, and either:

a. One additional shutdown cooling loop shall be OPERABLE#, or

b. The secondary side water level of at least two steam generators shall be greater than 25% indicated wide range level.

APPLICABILITY: MODE 5 with reactor coolant loops filled##.

ACTION:

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

b. With no shutdown cooling 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 shutdown cooling loop to operation.

SURVEILLANCE REQUIREMENTS

4.4.1.4.1.1 The secondary side water level of both steam generators when required shall be determined to be within limits at least once per 12 hours.

4.4.1.4.1.2 At least one shutdown cooling loop shall be determined to be in operation and circulating reactor coolant at a flow rate of greater than or equal to 4000 gpm at least once per 12 hours.

#One shutdown cooling loop may be inoperable for up to 2 hours for surveillance testing provided the other shutdown cooling loop is OPERABLE and in operation.

**

##A reactor coolant pump shall not be started with one or more of the Reactor Coolant System cold leg temperatures less than or equal to ~~[illegible]~~°F during cooldown, or ~~[illegible]~~°F during heatup, unless the secondary water temperature saturation temperature corresponding to steam generator pressure) of each steam generator is less than 100°F above each of the Reactor Coolant System cold leg temperatures.

*The shutdown cooling pump may be deenergized for up to 1 hour provided (1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

** See Applicant's SAR

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LIMITING CONDITION FOR OPERATION

3.4.1.4.2 Two shutdown cooling loops shall be OPERABLE# and at least one shutdown cooling loop shall be in operation.*

APPLICABILITY: MODE 5 with reactor coolant loops not filled.

ACTION:

a. With less than the above required loops OPERABLE, immediately initiate corrective action to return the required loops to OPERABLE status as soon as possible.

b. With no shutdown cooling 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 shutdown cooling loop to operation.

SURVEILLANCE REQUIREMENTS

4.4.1.4.2 At least one shutdown cooling loop shall be determined to be in operation and circulating reactor coolant at a flow rate of greater than or equal to 4000 gpm at least once per 12 hours.

#One shutdown cooling loop may be inoperable for up to 2 hours for surveillance testing provided the other shutdown cooling loop is OPERABLE and in operation.

*The shutdown cooling pump may be de-energized for up to 1 hour provided (1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

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3/4.4.2 SAFETY VALVES

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.2.1 A minimum of one pressurizer code safety valve shall be OPERABLE with a lift setting of 2500 psia ± 1%.*

APPLICABILITY: MODE 4.

ACTION:

a. With no pressurizer code safety valve OPERABLE, immediately suspend all operations involving positive reactivity changes and place an OPERABLE shutdown cooling loop into operation.

b. The provisions of Specification 3.0.4 may be suspended for up to 12 hours for entering into and during operation in MODE 4 for purposes of setting the pressurizer code safety valves under ambient (HOT) conditions provided a preliminary cold setting was made prior to heatup.

SURVEILLANCE REQUIREMENTS

4.4.2.1 No additional Surveillance Requirements other than those required by Specification 4.0.5.

*The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

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LIMITING CONDITION FOR OPERATION

3.4.2.2 All pressurizer code safety valves shall be OPERABLE with a lift setting of 2500 psia ± 1%.*

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

With one pressurizer code safety valve inoperable, either restore the inoperable valve to OPERABLE status within 15 minutes or be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours with the shutdown cooling system suction line relief valves aligned to provide overpressure protection for the Reactor Coolant System.

SURVEILLANCE REQUIREMENTS

4.4.2.2 No additional Surveillance Requirements other than those required by Specification 4.0.5.

*The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

3/4.4.3 PRESSURIZER

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PRESSURIZER

LIMITING CONDITION FOR OPERATION

Insert

3.4.3.1 The pressurizer shall be OPERABLE with: ~~a minimum steady-state water level of greater than or equal to 27% indicated level (425 cubic feet) and a maximum steady state water level of less than or equal to 56% indicated level (948 cubic feet) and at least two groups of pressurizer heaters capable of being powered from Class 1E buses each having a nominal capacity of at least 150 kW.~~

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With only one group of the above required pressurizer heaters OPERABLE, restore at least two groups to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With the pressurizer otherwise inoperable, restore the pressurizer to OPERABLE status within 1 hour, or be in at least HOT STANDBY with the reactor trip breakers open within 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.4.3.1.1 The pressurizer water volume shall be determined to be within its limits at least once per 12 hours.

* 4.4.3.1.2 The capacity of the above required groups of pressurizer heaters shall be verified to be at least 150 kW at least once per 92 days.

4.4.3.1.3 The emergency power supply for the pressurizer heaters shall be demonstrated OPERABLE at least once per 18 months by verifying that on an Engineered Safety Features Actuation test signal concurrent with a loss-of-offsite power:

a. The pressurizer heaters are automatically shed from the emergency power sources, and

b. The pressurizer heaters can be reconnected to their respective buses manually from the control room.

Insert

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a. A steady state water volume less than or equal to 58% indicated level (~~1010~~ cu. ft) but greater than 27% indicated level (44~~[illegible]~~ 6 cu. ft.), and 996

b. At least two groups of pressurizer heaters capable of being powered from 1E buses each having a nominal capacity of at least 150 kw.

AUXILIARY SPRAY

LIMITING CONDITION FOR OPERATION

3.4.3.2 Both auxiliary spray valves shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With only one of the above required auxiliary spray valves OPERABLE, restore both valves to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With none of the above required auxiliary spray valves OPERABLE, restore at least one valve to OPERABLE status within the next 6 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.4.3.2.1 The auxiliary spray valves shall be verified to have power available to each valve every 24 hours.

4.4.3.2.2 The auxiliary spray valves shall be cycled at least once per 18 months.

3/4.4.4 STEAM GENERATORS

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LIMITING CONDITION FOR OPERATION

3.4.4 Each steam generator shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With one or more steam generators inoperable, restore the inoperable generator(s) to OPERABLE status prior to increasing T_{cold} above 210°F.

SURVEILLANCE REQUIREMENTS

4.4.4.0 Each steam generator shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program.

4.4.4.1 Steam Generator Sample Selection and Inspection - Each steam generator shall be determined OPERABLE during shutdown by selecting and inspecting at least the minimum number of steam generators specified in Table 4.4-1.

4.4.4.2 Steam Generator Tube Sample Selection and Inspection - The steam generator tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Table 4.4-2. The inservice inspection of steam generator tubes shall be performed at the frequencies specified in Specification 4.4.4.3 and the inspected tubes shall be verified acceptable per the acceptance criteria of Specification 4.4.4.4. The tubes selected for each inservice inspection shall include at least 3% of the total number of tubes in all steam generators; the tubes selected for these inspections shall be selected on a random basis except:

a. Where experience in similar plants with similar water chemistry indicates critical areas to be inspected, then at least 50% of the tubes inspected shall be from these critical areas.

b. The first sample of tubes selected for each inservice inspection (subsequent to the preservice inspection) of each steam generator shall include:

SURVEILLANCE REQUIREMENTS (Continued)

1. All nonplugged tubes that previously had detectable wall penetrations (greater than 20%).

2. Tubes in those areas where experience has indicated potential problems.

3. A tube inspection (pursuant to Specification 4.4.4.4a.8.) shall be performed on each selected tube. If any selected tube does not permit the passage of the eddy current probe for a tube inspection, this shall be recorded and an adjacent tube shall be selected and subjected to a tube inspection.

c. The tubes selected as the second and third samples (if required by Table 4.4-2) during each inservice inspection may be subjected to a partial tube inspection provided:

1. The tubes selected for these samples include the tubes from those areas of the tube sheet array where tubes with imperfections were previously found.

2. The inspections include those portions of the tubes where imperfections were previously found.

The results of each sample inspection shall be classified into one of the following three categories:

Category	Inspection Results
C-1	Less than 5% of the total tubes inspected are degraded tubes and none of the inspected tubes are defective.
C-2	One or more tubes, but not more than 1% of the total tubes inspected are defective, or between 5% and 10% of the total tubes inspected are degraded tubes.
C-3	More than 10% of the total tubes inspected are degraded tubes or more than 1% of the inspected tubes are defective.

Note: In all inspections, previously degraded tubes must exhibit significant (greater than 10%) further wall penetrations to be included in the above percentage calculations.

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4.4.4.3 Inspection Frequencies - The above required inservice inspections of steam generator tubes shall be performed at the following frequencies:

a. The first inservice inspection shall be performed after 6 Effective Full Power Months but within 24 calender months of initial criticality. Subsequent inservice inspections shall be performed at intervals of not less than 12 nor more than 24 calendar months after the previous inspection. If two consecutive inspections following service under AVT conditions, not including the preservice inspection, result in all inspection results falling into the C-1 category or if two consecutive inspections demonstrate that previously observed degradation has not continued and no additional degradation has occurred, the inspection interval may be extended to a maximum of once per 40 months.

b. If the results of the inservice inspection of a steam generator conducted in accordance with Table 4.4-2 at 40 month intervals fall into Category C-3, the inspection frequency shall be increased to at least once per 20 months. The increase in inspection frequency shall apply until the subsequent inspections satisfy the criteria of Specification 4.4.4.3a.; the interval may then be extended to a maximum of once per 40 months.

c. Additional, unscheduled inservice inspections shall be performed on each steam generator in accordance with the first sample inspection specified in Table 4.4-2 during the shutdown subsequent to any of the following conditions:

   1. Primary-to-secondary tubes leaks (not including leaks originating from tube-to-tube sheet welds) in excess of the limits of Specification 3.4.5.2.

   2. A seismic occurrence greater than the Operating Basis Earthquake.

   3. A loss-of-coolant accident requiring actuation of the engineered safeguards.

   4. A main steam line or feedwater line break.

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4.4.4.4 Acceptance Criteria

a. As used in this Specification

1. Imperfection means an exception to the dimensions, finish, or contour of a tube from that required by fabrication drawings or specifications. Eddy-current testing indications below 20% of the nominal tube wall thickness, if detectable, may be considered as imperfections.

2. Degradation means a service-induced cracking, wastage, wear, or general corrosion occurring on either inside or outside of a tube.

3. Degraded Tube means a tube containing imperfections greater than or equal to 20% of the nominal wall thickness caused by degradation.

4. % Degradation means the percentage of the tube wall thickness affected or removed by degradation.

5. Defect means an imperfection of such severity that it exceeds the plugging limit. A tube containing a defect is defective.

6. Plugging Limit means the imperfection depth at or beyond which the tube shall be removed from service and is equal to ~~40%~~ ** of the nominal tube wall thickness.

7. Unserviceable describes the condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of an Operating Basis Earthquake, a loss-of-coolant accident, or a steam line or feedwater line break as specified in 4.4.4.3c., above.

8. Tube Inspection means an inspection of the steam generator tube from the point of entry (hot leg side) completely around the U-bend to the top support of the cold leg.

9. Preservice Inspection means an inspection of the full length of each tube in each steam generator performed by eddy current techniques prior to service to establish a baseline

** value to be determined in accordance with recommendations of Regulatory Guide 1.121, August 1976.

condition of the tubing. This inspection was performed prior to the field hydrostatic test and prior to initial POWER OPERATION using the equipment and techniques expected to be used during subsequent inservice inspections.

b. The steam generator shall be determined OPERABLE after completing the corresponding actions (plug all tubes exceeding the plugging limit and all tubes containing through-wall cracks) required by Table 4.4-2.

4.4.4.5 Reports

a. Within 15 days following the completion of each inservice inspection of steam generator tubes, the number of tubes plugged in each steam generator shall be reported to the Commission in a Special Report pursuant to Specification 6.9.2.

b. The complete results of the steam generator tube inservice inspection shall be submitted to the Commission in a Special Report pursuant to Specification 6.9.2 within 12 months following completion of the inspection. This Special Report shall include:

1. Number and extent of tubes inspected.

2. Location and percent of wall-thickness penetration for each indication of an imperfection.

3. Identification of tubes plugged.

c. Results of steam generator tube inspections which fall into Category C-3 shall be reported in a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days and prior to resumption of plant operation and shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

TABLE 4.4-1

MINIMUM NUMBER OF STEAM GENERATORS TO BE

INSPECTED DURING INSERVICE INSPECTION

The inservice inspection may be limited to one steam generator on a rotating schedule encompassing 6% of the tubes if the results of the first or previous inspections indicate that all steam generators are performing in a like manner. Note that under some circumstances, the operating conditions in one or more steam generators may be found to be more severe than those in other steam generators. Under such circumstances the sample sequence shall be modified to inspect the most severe conditions.

TABLE 4.4-2

STEAM GENERATOR TUBE INSPECTION

1ST SAMPLE INSPECTION			2ND SAMPLE INSPECTION		3RD SAMPLE INSPECTION	
Sample Size	Result	Action Required	Result	Action Required	Result	Action Required
A minimum of S Tubes per S. G.	C-1	None	N. A.	N. A.	N. A.	N. A.
	C-2	Plug defective tubes and inspect additional 2S tubes in this S. G.	C-1	None	N. A.	N. A.
			C-2	Plug defective tubes and inspect additional 4S tubes in this S. G.	C-1	None
					C-2	Plug defective tubes
					C-3	Perform action for C-3 result of first sample
			C-3	Perform action for C-3 result of first sample	N. A.	N. A.
	C-3	Inspect all tubes in this S. G., plug defective tubes and inspect 2S tubes in each other S. G. Notification to NRC pursuant to §50.72 (b)(2) of 10 CFR Part 50	All other S. G.s are C-1	None	N. A.	N. A.
			Some S. G.s C-2 but no additional S. G. are C-3	Perform action for C-2 result of second sample	N. A.	N. A.
			Additional S. G. is C-3	Inspect all tubes in each S. G. and plug defective tubes. Notification to NRC pursuant to §50.72 (b)(2) of 10 CFR Part 50	N. A.	N. A.

$S = 3 \frac{N}{n}\%$ Where N is the number of steam generators in the unit, and n is the number of steam generators inspected during an inspection

3/4.4.5 REACTOR COOLANT SYSTEM LEAKAGE

LEAKAGE DETECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.5.1 The following Reactor Coolant System leakage detection systems shall be OPERABLE:

a. A containment atmosphere particulate radioactivity monitoring system,

b. The containment sump level and flow monitoring system, and

c. The containment atmosphere gaseous radioactivity monitoring system.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With only two of the above required leakage detection systems OPERABLE, operation may continue for up to 30 days provided grab samples of the containment atmosphere are obtained and analyzed at least once per 24 hours when the required gaseous and/or particulate radioactivity monitoring system is inoperable; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.5.1 The leakage detection systems shall be demonstrated OPERABLE by:

a. Containment atmosphere gaseous and particulate monitoring system-performance of CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST at the frequencies specified in Table 4.3-3,

b. Containment sump level and flow monitoring system-performance of CHANNEL CALIBRATION at least once per 18 months.

3.4.5.1 Reactor Coolant System Leakage Detection Systems

See Applicant's SAR

OPERATIONAL LEAKAGE

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LIMITING CONDITION FOR OPERATION

3.4.5.2 Reactor Coolant System leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE,

b. 1 gpm UNIDENTIFIED LEAKAGE, (*)

c. 1 gpm total primary-to-secondary leakage through all steam generators, and ~~720~~ gallons per day through any one steam generator,

d. 10 gpm IDENTIFIED LEAKAGE from the Reactor Coolant System, and

e. 1 gpm leakage at a Reactor Coolant System pressure of 2250 ± 20 psia from any Reactor Coolant System pressure isolation valve specified in Table 3.4-1.

APPLICABILITY: MODES 1, 2, 3, and 4

ACTION:

a. With any PRESSURE BOUNDARY LEAKAGE, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With any Reactor Coolant System leakage greater than any one of the limits, excluding PRESSURE BOUNDARY LEAKAGE and leakage from Reactor Coolant System pressure isolation valves, reduce the leakage rate to within its limits within 4 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With any Reactor Coolant System pressure isolation valve leakage greater than the above limit, isolate the high pressure portion of the affected system from the low pressure portion within 4 hours by use of at least one closed manual or deactivated automatic valve, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

* d. With RCS leakage alarmed and confirmed in a flow path with no flow rate indicators, commence an RCS water inventory balance within 1 hour to determine the leak rate.

SURVEILLANCE REQUIREMENTS

4.4.5.2.1 Reactor Coolant System leakages shall be demonstrated to be within each of the above limits by:

a. Monitoring the containment atmosphere gaseous and particulate radioactivity monitor at least once per 12 hours.

b. Monitoring the containment sump inventory and discharge at least once per 12 hours.

* See Applicant's SAR

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SURVEILLANCE REQUIREMENTS (Continued)

c. Performance of a Reactor Coolant System water inventory balance at least once per 72 hours.

d. Monitoring the reactor head flange leakoff system at least once per 24 hours.

4.4.5.2.2 Each Reactor Coolant System pressure isolation valve specified in Table 3.4-1 shall be demonstrated OPERABLE by verifying leakage to be within its limit:

a. At least once per 18 months,

b.* Prior to entering MODE 2 whenever the plant has been in COLD SHUTDOWN for 72 hours or more and if leakage testing has not been performed in the previous 9 months,

c. Prior to returning the valve to service following maintenance, repair or replacement work on the valve,

d.* Within 24 hours following valve actuation due to automatic or manual action or flow through the valve,

~~e.* Within 72 hours following a system response to an Engineered Safety Feature actuation signal.~~

The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 or 4.

~~*The provisions of Specifications 4.4.5.2.2.b, 4.4.5.2.2.d, and 4.4.5.2.2.e are not applicable for valves UV 651, UV 652, UV 653 and UV 654 due to position indication of valves in the control room.~~

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TABLE 3.4-1

REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVES

VALVE [illegible]	DESCRIPTION
1) [illegible]	~~LOOP 1A RC/SI CHECK~~
2) [illegible]	~~LOOP 1B RC/SI CHECK~~
3) [illegible]	~~LOOP 2A RC/SI CHECK~~
4) [illegible]	~~LOOP 2B RC/SI CHECK~~
5) [illegible]	~~LOOP 1A SIT CHECK~~
6) [illegible]	~~LOOP 1B SIT CHECK~~
7) [illegible]	~~LOOP 2A SIT CHECK~~
8) [illegible]	~~LOOP 2B SIT CHECK~~
9) [illegible]	~~LOOP 1A SI HEADER CHECK~~
10) [illegible]	~~LOOP 2B SI HEADER CHECK~~
~~11) SIE-V540~~	~~LOOP 2A SI HEADER CHECK~~
~~12) SIE-V541~~	~~LOOP 2B SI HEADER CHECK~~
~~13) SIA-V522~~	~~LOOP 1 HP LONG TERM RECIRCULATION CHECK~~
~~14) SIA-V523~~	~~LOOP 1 HP LONG TERM RECIRCULATION CHECK~~
~~15) SIB-V532~~	~~LOOP 2 HP LONG TERM RECIRCULATION CHECK~~
~~16) SIB-V533~~	~~LOOP 2 HP LONG TERM RECIRCULATION CHECK~~
~~17) SIA-UV651*,#~~	~~LOOP 1 SHUTDOWN COOLING ISOLATION~~
~~18) SIB-UV652*,#~~	~~LOOP 2 SHUTDOWN COOLING ISOLATION~~
~~19) SIC-UV653*,#~~	~~LOOP 1 SHUTDOWN COOLING ISOLATION~~
~~20) SID-UV654*,#~~	~~LOOP 2 SHUTDOWN COOLING ISOLATION~~

~~*Testing per Specification 4.4.5.2.2.d is not applicable due to positive indication of valve position in the control room.~~

~~#1. Leakage rates greater than 1.0 gpm but less than or equal to 5.0 gpm are considered acceptable if the latest measured rate has not exceeded the rate determined by previous test by an amount that reduces the margin between measured leakage rate and the maximum permissible rate of 5.0 gpm by 50% or greater.~~

~~2. Leakage rates greater than 1.0 gpm but less than or equal to 5.0 gpm are considered unacceptable if the latest measured rate exceeded the rate determined by the previous test by an amount that reduces the margin between measured leakage rate and the maximum permissible rate of 5.0 gpm by 50% or greater.~~

~~3. Leakage rates greater than 5.0 gpm are considered unacceptable.~~

* See Applicant's SAR

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3/4.4.6 CHEMISTRY

LIMITING CONDITION FOR OPERATION

3.4.6 The Reactor Coolant System chemistry shall be maintained within the limits specified in Table 3.4-2.

APPLICABILITY: At all times.

ACTION:

MODES 1, 2, 3, and 4:

a. With any one or more chemistry parameter in excess of its Steady State Limit but within its Transient Limit, restore the parameter to within its Steady State Limit within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With any one or more chemistry parameter in excess of its Transient Limit, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

At All Other Times:

With the concentration of either chloride or fluoride in the Reactor Coolant System in excess of its Steady State Limit for more than 24 hours or in excess of its Transient Limit, reduce the pressurizer pressure to less than or equal to 500 psia, if applicable, and perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the Reactor Coolant System; determine that the Reactor Coolant System remains acceptable for continued operation prior to increasing the pressurizer pressure above 500 psia or prior to proceeding to MODE 4.

SURVEILLANCE REQUIREMENTS

4.4.6 The Reactor Coolant System chemistry shall be determined to be within the limits by analysis of those parameters at the frequencies specified in Table 4.4-3.

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TABLE 3.4-2

REACTOR COOLANT SYSTEM

CHEMISTRY

PARAMETER	STEADY STATE LIMIT	TRANSIENT LIMIT
DISSOLVED OXYGEN*	≤ 0.10 ppm	≤ 1.00 ppm
CHLORIDE	≤ 0.15 ppm	≤ 1.50 ppm
FLUORIDE	≤ 0.10 ppm	≤ 1.00 ppm

*Limit not applicable with T_{cold} less than or equal to 250°F.

TABLE 4.4-3

REACTOR COOLANT SYSTEM

CHEMISTRY LIMITS SURVEILLANCE REQUIREMENTS

PARAMETER	SAMPLE AND ANALYSIS FREQUENCY
DISSOLVED OXYGEN*	At least once per 72 hours
CHLORIDE	At least once per 72 hours
FLUORIDE	At least once per 72 hours

*Not required with T_{cold} less than or equal to 250°F

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3/4.4.7 SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

3.4.7 The specific activity of the primary coolant shall be limited to:

a. Less than or equal to 1.0 microcurie/gram DOSE EQUIVALENT I-131, and

b. Less than or equal to $100/\bar{E}$ microcuries/gram.

APPLICABILITY: MODES 1, 2, 3, 4, and 5.

ACTION:

MODES 1, 2, and 3*:

a. With the specific activity of the primary coolant greater than 1.0 microcurie/gram DOSE EQUIVALENT I-131 but within the allowable limit (below and to the left of the line) shown on Figure 3.4-1, operation may continue for up to 48 hours provided that the cumulative operating time under these circumstances does not exceed 800 hours in any consecutive 12 month period. With the total cumulative operating time at a primary coolant specific activity greater than 1.0 microcurie/gram DOSE EQUIVALENT I-131 exceeding 500 hours in any consecutive 6 month period, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days indicating the number of hours above this limit. The provisions of Specification 3.0.4 are not applicable.

b. With the specific activity of the primary coolant greater than 1.0 microcurie/gram DOSE EQUIVALENT I-131 for more than 48 hours during one continuous time interval or exceeding the limit line shown on Figure 3.4-1, be in at least HOT STANDBY with T_{cold} less than 500°F within 6 hours.

c. With the specific activity of the primary coolant greater than $100/\bar{E}$ microcuries/gram, be in at least HOT STANDBY with T_{cold} less than 500°F within 6 hours.

*With T_{cold} greater than or equal to 500°F.

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LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

MODES 1, 2, 3, 4, and 5:

d. With the specific activity of the primary coolant greater than 1 microcurie/gram DOSE EQUIVALENT I-131 or greater than 100/Ē microcuries/gram, perform the sampling and analysis requirements of item 4.(a) of Table 4.4-4 until the specific activity of the primary coolant is restored to within its limits. A Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 ~~within 30 days with a copy to the Director, Nuclear Reactor Regulation, Attention: Chief, Core Performance Branch, and Chief, Accident Evaluation Branch, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555.~~ This report shall contain the results of the specific activity analyses together with the following information:

of Administrative Controls

1. Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded,

2. Fuel burnup by core region,

3. Clean-up flow history starting 48 hours prior to the first sample in which the limit was exceeded.

4. History of degassing operation, if any, starting 48 hours prior to the first sample in which the limit was exceeded, and

5. The time duration when the specific activity of the primary coolant exceeded 1 microcurie/gram DOSE EQUIVALENT I-131.

SURVEILLANCE REQUIREMENTS

4.4.7 The specific activity of the primary coolant shall be determined to be within the limits by performance of the sampling and analysis program of Table 4.4-4.

~~PALO VERDE - UNIT 1~~ CESSAR80-NSSS-STS

TABLE 4.4-4

PRIMARY COOLANT SPECIFIC ACTIVITY SAMPLE AND ANALYSIS PROGRAM

TYPE OF MEASUREMENT AND ANALYSIS	SAMPLE AND ANALYSIS FREQUENCY	MODES IN WHICH SAMPLE AND ANALYSIS REQUIRED
1. Gross Activity Determination	At least once per 72 hours	1, 2, 3, 4
2. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	1 per 14 days	1
3. Radiochemical for $\bar{E}$ Determination	1 per 6 months*	1
4. Isotopic Analysis for Iodine Including I-131, I-133, and I-135	(a) Once per 4 hours, whenever the specific activity exceeds 1.0 μCi/gram, DOSE EQUIVALENT I-131 or 100/$\bar{E}$ μCi/gram, and	1#, 2#, 3#, 4#, 5#
	(b) One sample between 2 and 6 hours following a THERMAL POWER change exceeding 15% of the RATED THERMAL POWER within a 1-hour period. One sample is sufficient if plant has gone through a SHUTDOWN or if transient is complete in 6 hours.	1, 2, 3

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Until the specific activity of the primary coolant system is restored within its limits.

* Sample to be taken after a minimum of 2 EFPD and 20 days of POWER OPERATION have elapsed since reactor was last subcritical for 48 hours or longer.

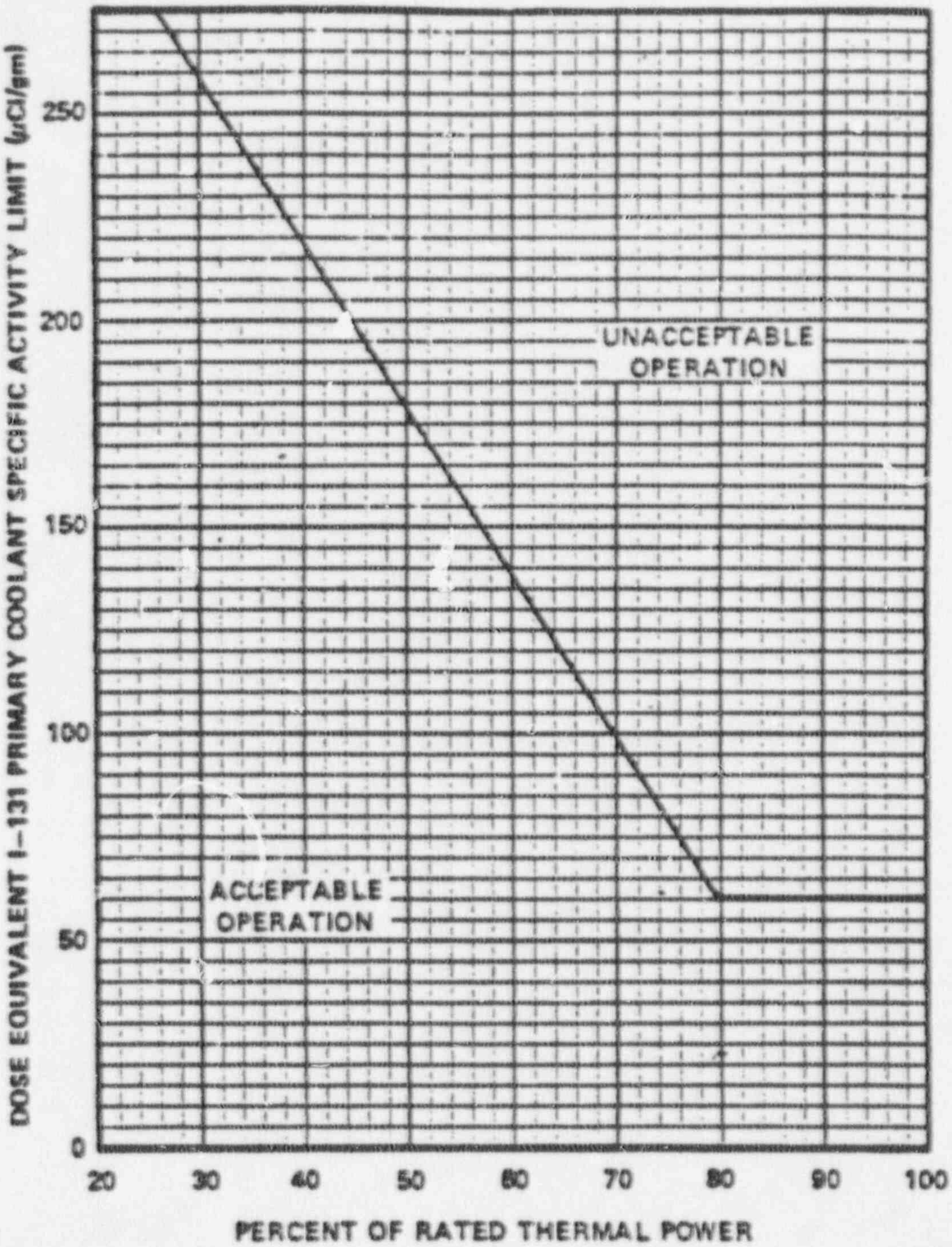

FIGURE 3.4-1

DOSE EQUIVALENT I-131 PRIMARY COOLANT SPECIFIC ACTIVITY LIMIT VERSUS PERCENT OF RATED THERMAL POWER WITH THE PRIMARY COOLANT SPECIFIC ACTIVITY > 1.0 μCi/GRAM DOSE EQUIVALENT I-131

3/4.4.8 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

3.4.8.1 The Reactor Coolant System (except the pressurizer) temperature and pressure shall be limited in accordance with the limit lines shown on Figure 3.4-2 during heatup, cooldown, criticality, and inservice leak and hydrostatic testing with:

a. A maximum heatup rate of 100°F per hour. ~~with the RCS cold leg temperature less than or equal to 95°F, 40°F per hour with RCS cold leg temperature greater than 95°F but less than or equal to 400°F, and 100°F per hour with RCS cold leg temperature greater than 400°F~~.

b. A maximum cooldown rate of 100°F per hour. ~~with RCS cold leg temperature less than or equal to 100°F, 40°F per hour with RCS cold leg temperature greater than 100°F but less than or equal to 120°F, and 100°F per hour with RCS cold leg temperature greater than 120°F~~.

c. A maximum temperature change of 10°F in any 1-hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves on Figure 3.4-2.

APPLICABILITY: At all times.*

ACTION:

With any of the above limits exceeded, restore the temperature and/or pressure to within the limit within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the Reactor Coolant System; determine that the Reactor Coolant System remains acceptable for continued operations or be in at least HOT STANDBY within the next 6 hours and reduce the RCS T_{cold} and pressure to less than 210°F and 500 psia, respectively, within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.8.1.1 The Reactor Coolant System temperature and pressure shall be determined to be within the limits at least once per 30 minutes during system heatup, cooldown, and inservice leak and hydrostatic testing operations.

4.4.8.1.2 The reactor vessel material irradiation surveillance specimens shall be removed and examined, to determine changes in material properties, at the intervals required by 10 CFR Part 50 Appendix H in accordance with the schedule in Table 4.4-5. The results of these examinations shall be used to update Figure 3.4-2.

*See Special Test Exception 3.10.5.

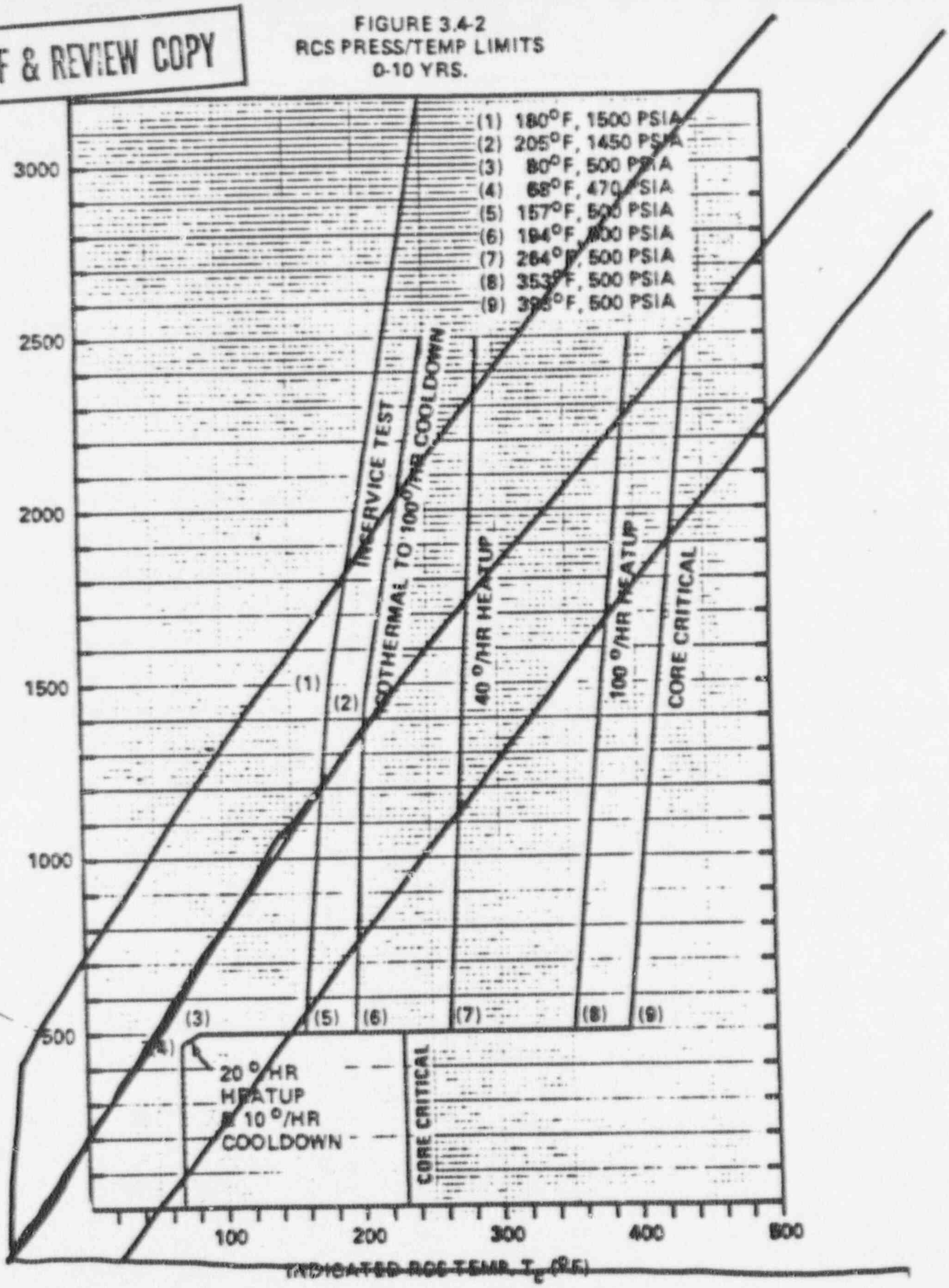

FIGURE 3/4 3.4-2

RCS PRESS/TEMP LIMITS (0 - 10 YRS) FULL POWER OPERATION
(SEE APPLICANT'S FSAR)

TABLE 4.4-5

REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM - WITHDRAWAL SCHEDULE *

CAPSULE NUMER	VESSEL LOCATION	LEAD FACTOR	WITHDRAWAL TIME (EFPY)
1	38°	1.5	~~0 - 10~~ Standby
2	43°	1.5	Standby
3	137°	1.5	4 - 5
4	142°	1.5	Standby
5	230°	1.5	12 - 15
6	310°	1.5	18 - 24

* Typical

PRESSURIZER HEATUP/COOLDOWN LIMITS

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LIMITING CONDITION FOR OPERATION

3.4.8.2 The pressurizer temperature shall be limited to:

a. A maximum heatup rate of 200°F per hour, and

b. A maximum cooldown rate of 200°F per hour.

APPLICABILITY: At all times.

ACTION:

With the pressurizer temperature limits in excess of any of the above limits, restore the temperature to within the limits within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the pressurizer; determine that the pressurizer remains acceptable for continued operation or be in at least HOT STANDBY within the next 6 hours and reduce the pressurizer pressure to less than 500 psig within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.8.2.1 The pressurizer temperatures shall be determined to be within the limits at least once per 30 minutes during system heatup or cooldown.

4.4.8.2.2 The spray water temperature differential shall be determined for use in Table 5.7-2 for each cycle of main spray with less than four reactor coolant pumps operating and for each cycle of auxiliary spray operation.

**

LIMITING CONDITION FOR OPERATION

3.4.8.3 Both shutdown cooling system (SCS) suction line relief valves with lift settings of less than or equal to ~~[illegible]~~ psig shall be OPERABLE and aligned to provide overpressure protection for the Reactor Coolant System.

APPLICABILITY: When the reactor vessel head is installed and the temperature of one or more of the RCS cold legs is less than or equal to:

a. ** ~~255~~°F during cooldown

b. ** ~~305~~°F during heatup

ACTION:

a. With one SCS relief valve inoperable, restore the inoperable valve to OPERABLE status within seven days or reduce T_{cold} to less than 200°F and, depressurize and vent the RCS through a greater than or equal to ** ~~X~~ square inch vent(s) within the next eight hours. Do not start a reactor coolant pump if the steam generator secondary water temperature is greater than 100°F above any RCS cold leg temperature.

b. With both SCS relief valves inoperable, reduce T_{cold} to less than 200°F and, depressurize and vent the RCS through a greater than or equal to ** ~~X~~ square inch vent(s) within eight hours. Do not start a reactor coolant pump if the steam generator secondary water temperature is greater than 100°F above any RCS cold leg temperature.

c. In the event either the SCS suction line relief valves or an RCS vent(s) are used to mitigate an RCS pressure transient, a Special Report shall be prepared and submitted to the Commission pursuant to ~~Specification 6.9.2~~ Administrative Controls within 30 days. The report shall describe the circumstances initiating the transient, the effect of the SCS suction line relief valves or RCS vent(s) on the transient and any corrective action necessary to prevent recurrence.

d. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.8.3.1 Each SCS suction line relief valve shall be verified to be aligned to provide overpressure protection for the RCS once every 8 hours during

a. Cooldown with the RCS temperature less than or equal to ~~255~~°F. **

b. Heatup with the RCS temperature less than or equal to ~~305~~°F. **

4.4.8.3.2 The SCS suction line relief valves shall be verified OPERABLE with the required setpoint at least once per 18 months.

** See Applicant's SAR

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3/4.4.9 STRUCTURAL INTEGRITY

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LIMITING CONDITION FOR OPERATION

3.4.9 The structural integrity of ASME Code Class 1, 2, and 3 components shall be maintained in accordance with Specification 4.4.9.

APPLICABILITY: ALL MODES

ACTION:

a. With the structural integrity of any ASME Code Class 1 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature more than 50°F above the minimum temperature required by NDT considerations.

b. With the structural integrity of any ASME Code Class 2 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature above 210°F.

c. With the structural integrity of any ASME Code Class 3 component(s) not conforming to the above requirements, restore the structural integrity of the affected component to within its limit or isolate the affected component from service.

d. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.9 In addition to the requirements of Specification 4.0.5, each reactor coolant pump flywheel shall be inspected per the recommendations of Regulatory Position C.4.b of Regulatory Guide 1.14, Revision ~~1~~, ~~October 27, 1971~~.

** **

** See Applicant's SAR for revision number and date of the Regulatory Guide

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3/4.4.10 REACTOR COOLANT SYSTEM VENTS

LIMITING CONDITION FOR OPERATION

3.4.10 Both reactor coolant system vent paths from the reactor vessel head shall be OPERABLE and closed.

APPLICABILITY: MODES 1, 2, 3 and 4

ACTION:

a. With only one of the above required reactor vessel head vent paths OPERABLE, restore both paths to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in ~~HOT~~ COLD SHUTDOWN within the following ~~6~~ 30 hours.

b. With none of the above required reactor vessel head vent paths OPERABLE, restore at least one path to OPERABLE status within the next 6 hours or be in at least HOT STANDBY within the next 6 hours and in ~~HOT~~ COLD SHUTDOWN within the following ~~6~~ 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.10 Each Reactor Coolant System vent path shall be demonstrated OPERABLE at least once per 18 months [redacted] by:

a. Verifying all manual isolation valves in each vent path are locked in the open position.

b. Cycling each vent through at least one complete cycle from the control room.

c. Verifying flow through the reactor coolant system vent paths during venting.

3/4.5.1 SAFETY INJECTION TANKS

LIMITING CONDITION FOR OPERATION

3.5.1 Each Reactor Coolant System safety injection tank shall be OPERABLE with:

a. The isolation valve key-locked open and power to the valve removed,

b. A contained borated water level of between 28% ~~(1802 cubic feet)~~ and 72% ~~(1914 cubic feet)~~

c. A boron concentration between ~~2000~~ 4000 and 4400 ppm of boron, and

d. A nitrogen cover-pressure of between 600 and 625 psig.

e. Nitrogen vent valves closed and power removed.**

f. Nitrogen vent valves are capable of being operated upon restoration of power.

APPLICABILITY: MODES 1*, 2*, 3,*†, and 4*†.

ACTION:

a. With one safety injection tank inoperable, except as a result of a closed isolation valve, restore the inoperable tank to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With one safety injection tank inoperable due to the isolation valve being closed, either immediately open the isolation valve or be in at least HOT STANDBY within 1 hour and be in HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.5.1 Each safety injection tank shall be demonstrated OPERABLE:

a. At least once per 12 hours by:

1. Verifying the contained borated water volume and nitrogen cover-pressure in the tanks is within the above limits, and

†With pressurizer pressure greater than or equal to ~~2037~~ 1750 psia. When pressurizer pressure is less than ~~2037~~ 1750 psia, at least three safety injection tanks must be OPERABLE, each with a minimum pressure of 254 psig and a maximum pressure of 625 psig, and a contained borated water volume of between 0% narrow range (corresponding to 60% wide range indication or 1415 cubic feet) and 72% narrow range indication (corresponding to 81% wide range indication or 1914 cubic feet). With all four safety injection tanks OPERABLE, each tank shall have a minimum pressure of 254 psig and a maximum pressure of 625 psig, and a contained borated water volume of between 0% narrow range (corresponding to 39% wide range indication or 962 cubic feet) and 72% narrow range indication (corresponding to 81% wide range indication or 1914 cubic feet). In MODE 4 with pressurizer pressure less than 430 psia, the safety injection tanks may be isolated.

*See Special Test Exceptions ~~[illegible]~~ 3.10.7

**Nitrogen vent valves may be cycled as necessary to maintain the required nitrogen cover pressure per Specification 3.5.1d.

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SURVEILLANCE REQUIREMENTS (Continued)

2. Verifying that each safety injection tank isolation valve is open and the nitrogen vent valves are closed.

b. At least once per 31 days and within 6 hours after each solution level increase of greater than or equal to 7% of tank narrow range level by verifying the boron concentration of the safety injection tank solution is between ~~2000~~ ≥4000 and 4400 ppm.

c. At least once per 31 days when the RCS pressure is above ~~700~~ 715 psi~~g~~a, by verifying that power to the isolation valve operator is removed.

d. At least once per 18 months by verifying that each safety injection tank isolation valve opens automatically under each of the following conditions:

1. When an actual or simulated RCS pressure signal exceeds 515 psia, and

2. Upon receipt of a safety injection actuation (SIAS) test signal.

e. At least once per 18 months by verifying OPERABILITY of the RCS-SIT differential pressure alarm by simulating RCS pressure > 715 psia with SIT pressure < 600 psig.

f. At least once per 18 months, when SITs are isolated, by verifying the SIT nitrogen vent valves can be opened.

g. At least once per 31 days, by verifying that power is removed from the nitrogen vent valves.

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3/4.5.2 ECCS SUBSYSTEMS - T_{cold} GREATER THAN OR EQUAL TO 350°F

LIMITING CONDITION FOR OPERATION

3.5.2 Two independent Emergency Core Cooling System (ECCS) subsystems shall be OPERABLE with each subsystem comprised of:

a. One OPERABLE high-pressure safety injection pump,

b. One OPERABLE low-pressure safety injection pump, and

c. An independent OPERABLE flow path capable of taking suction from the refueling water tank on a safety injection actuation signal and automatically transferring suction to the containment sump on a recirculation actuation signal.

APPLICABILITY: MODES 1, 2, and 3*.

ACTION:

a. With one ECCS subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 of Administrative Controls within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date. The current value of the usage factor for each affected injection nozzle shall be provided in this Special Report whenever its value exceeds 0.70.

*With pressurizer pressure greater than or equal to ~~[illegible]~~ 1750 psia.

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operators removed:

SURVEILLANCE REQUIREMENTS

4.5.2 Each ECCS subsystem shall be demonstrated OPERABLE:

a. At least once per 12 hours by verifying that the following valves are in the indicated positions with the valve ~~key locked shut~~:

Valve Number	Valve Function	Valve Position
1. SI [redacted] 604	1. HOT LEG INJECTION	1. SHUT
~~2. SIC-HV-321~~	~~2. HOT LEG INJECTION~~	~~2. SHUT~~
2 ~~3~~. SI [redacted] 609	2 ~~3~~. HOT LEG INJECTION	2 ~~3~~. SHUT
~~4. SIB-HV-331~~	~~4. HOT LEG INJECTION~~	~~4. SHUT~~

b. At least once per 31 days by:

1. Verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position, and

2. Verifying that the ECCS piping is full of water by venting the ECCS pump casing and accessible discharge piping high points.

c. By a visual inspection which verifies that no loose debris (rags, trash, clothing, etc.) is present in the containment which could be transported to the containment sump and cause restriction of the pump suctions during LOCA conditions. This visual inspection shall be performed:

1. For all accessible areas of the containment prior to establishing CONTAINMENT INTEGRITY, and

2. For all the affected areas within containment at the completion of containment entry when CONTAINMENT INTEGRITY is established.

d. At least once per 18 months by:

** requirement applicable only if SI 604, SI 609, SI 321, SI 331 are not each supplied by an independent and redundant emergency power source.

1. A visual inspection of the containment sump and verifying that the subsystem suction inlets are not restricted by debris and that the sump components (trash racks, screens, etc.) show no evidence of structural distress or corrosion.

2. Verifying that a minimum total of ** ~~464 cubic feet of solid granular trisodium phosphate dodecahydrate (TSP) is contained within the TSP storage baskets.~~ (BOP)

~~3. Verifying that when a representative sample of 0.055 ± 0.001 lb of TSP from a TSP storage basket is submerged, without agitation, in 1.0 ± 0.05 gallons of 77 ± 9 °F borated water from the RWT, the pH of the mixed solution is raised to greater than or equal to 7 within 4 hours.~~

e. At least once per 18 months, during shutdown, by:

1. Verifying that each automatic valve in the flow path actuates to its correct position on an SIAS and RAS test signal.

2. Verifying that each of the following pumps start automatically upon receipt of a safety injection actuation test signal:

   a. High pressure safety injection pump.

   b. Low pressure safety injection pump.

3. Verifying that on a recirculation actuation test signal, the containment sump isolation valves open, the HPSI, LPSI and CS pump minimum bypass recirculation flow line isolation valves and combined SI mini-flow valve close, and the LPSI pumps stop.

f. By verifying that each of the following pumps develops the indicated differential pressure at or greater than their respective minimum allowable recirculation flow when tested pursuant to Specification 4.0.5:

1. High pressure safety injection pump greater than or equal to 1761 psid.

2. Low pressure safety injection pump greater than or equal to 165 psid.

** See Applicants SAR for means to control pH in the containment sump water after an LOCA.

SURVEILLANCE REQUIREMENTS (Continued)

g. By verifying the correct position of each electrical and/or mechanical position stop for the following ECCS throttle valves:

   1. Within 4 hours following completion of each valve stroking operation or maintenance on the valve when the ECCS subsystems are required to be OPERABLE.

   2. At least once per 18 months.

Insert

LPSI System Valve Number	Hot Leg Injection Valve Number
1. SI[redacted] 615, SI[redacted] 306	1. SI[redacted] 604
2. SI[redacted] 625, SI[redacted] 307	2. SI[redacted] 609
3. SI[redacted] 635	3. SI[redacted] 321
4. SI[redacted] 645	4. SI[redacted] 331

h. By performing a flow balance test, during shutdown, following completion of modifications to the ECCS subsystems that alter the subsystem flow characteristics and verifying the following flow rates:

HPSI System - Single Pump

The sum of the injection line flow rates, excluding the highest flow rate, is greater than or equal to [redacted] gpm. **

LPSI System - Single Pump

1. Injection Loop 1, total flow equal to ~~4000 ± 100~~ gpm **
2. Injection Legs 1A and 1B when tested individually, with the other leg isolated, shall be within ~~100~~ gpm of each other. **
3. Injection Loop 2, total flow equal to ~~4000 ± 100~~ gpm **
4. Injection Legs 2A and 2B when tested individually, with the other leg isolated, shall be within ~~100~~ gpm of each other. **

Simultaneous Hot Leg and Cold Leg Injection - Single Pump

1. Hot Leg, flow equal to [redacted] gpm **
2. Cold Leg, flow equal to [redacted] gpm **

** See Applicant's SAR

Insert

HPSI System
Valve Number

1. SI-617, SI-616
2. SI-627, SI-626
3. SI-637, SI-636
4. SI-647, SI-646

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3/4.5.3 ECCS UBSYSTEMS - T_{cold} LESS THAN 350°F

LIMITING CONDITION FOR OPERATION

3.5.3 As a minimum, one ECCS subsystem comprised of the following shall be OPERABLE:

a. An OPERABLE high pressure safety injection pump, and

b. An OPERABLE flow path capable of taking suction from the refueling water tank on a safety injection actuation signal and automatically transferring suction to the containment sump on a recirculation actuation signal.

APPLICABILITY: ~~MODE~~ MODES 3* and 4.

ACTION:

a. With no ECCS subsystem OPERABLE, restore at least one ECCS subsystem to OPERABLE status within 1 hour or be in COLD SHUTDOWN within the next 20 hours.

b. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date. The current value of the usage factor for each affected safety injection nozzle shall be provided in this Special Report whenever its value exceeds 0.70.

SURVEILLANCE REQUIREMENTS

4.5.3 The ECCS subsystem shall be demonstrated OPERABLE per the applicable surveillance requirements of Specification 4.5.2.

* With pressurizer pressure less than 1750 psia.

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3/4.5.4 REFUELING WATER TANK

LIMITING CONDITION FOR OPERATION

3.5.4 The refueling water tank (RWT) shall be OPERABLE with:

a. A minimum borated water volume as specified in Figure 3.1-2 of Specification 3.1.2.5, ~~and~~

b. A boron concentration between 4000 and 4400 ppm of boron, and

c. A solution temperature between 60°F and 120°F.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With the refueling water tank inoperable, restore the tank to OPERABLE status within 1 hour or be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.5.4 The RWT shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1. Verifying the contained borated water volume in the tank, and

2. Verifying the boron concentration of the water.

b. At least once per 24 hours by verifying the RWT temperature when the (outside) air temperature is outside the 60°F to 120°F range.

3/4.6 CONTAINMENT SYSTEMS

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3/4.6.1 PRIMARY CONTAINMENT

CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION AND SURVEILLANCE REQUIREMENTS

3.6.1.1 Primary CONTAINMENT INTEGRITY shall be maintained.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

Without primary CONTAINMENT INTEGRITY, restore CONTAINMENT INTEGRITY within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

INSERT 1

4.6.1.1 Primary CONTAINMENT INTEGRITY shall be demonstrated:

a. At least once per 31 days by verifying that all penetrations* not capable of being closed by OPERABLE containment automatic isolation valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves secured in their positions except as provided in Table 3.6-1 of Specification 3.6.3.

b. By verifying that each containment air lock is in compliance with the requirements of Specification 3.6.1.3.

c. After each closing of each penetration subject to Type B testing, except containment air locks, if opened following a Type A or B test, by leak rate testing the seal with gas at P_a 49.2 psig and verifying that when the measured leakage rate for these seals is added to the leakage rates determined pursuant to Specification 4.6.1.2d. for all other Type B and C penetrations, the combined leakage rate is less than or equal to 0.60 L_a.

---

*Except valves, blind flanges, and deactivated automatic valves which are located inside the containment and are locked, sealed, or otherwise secured in the closed position. These penetrations shall be verified closed during each COLD SHUTDOWN except that such verification need not be performed more often than once per 92 days.

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Insert

(Containment Isolation Valve and channel Weld Pressurization Systems (Optional)

3/4.6.1.1 Containment Integrity

3/4.6.1.2 Containment Leakage

3/4.6.1.3 Containment Air Locks

3/4.6.1.4

3/4.6.1.5 Internal Pressure

3/4.6.1.6 Air Temperature

3/4.6.1.7 Containment Vessel Structural Integrity

Table 4.6-1, Tendon Surveillance-First Year

Table 4.6-2, Tendon Lift-Off Force First Year

3/4.6.1.8 Containment Ventilation System

See Applicant's SAR

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3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

CONTAINMENT SPRAY SYSTEM (Credit taken for Iodine removal)

LIMITING CONDITION FOR OPERATION

3.6.2.1 Two independent containment spray systems shall be OPERABLE with each spray system capable of taking suction from the RWT on a containment spray actuation signal and automatically transferring suction to the containment sump on a recirculation actuation signal. Each spray system flow path from the containment sump shall be w[illegible] an OPERABLE shutdown cooling heat exchanger.

APPLICABILITY: M[illegible]S 1, 2, 3, and 4.

ACTION:

With one containment spray system inoperable, restore the inoperable spray system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours; restore the inoperable spray system to OPERABLE status within the next 48 hours or be in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.1 Each containment spray system shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) in the flow path is positioned to take suction from the RWT on a containment spray actuation (CSAS) test signal.

b. By verifying that each pump develops an indicated differential pressure of greater than or equal to ~~[illegible]~~ 200 psid at greater than or equal the minimum allowable bypass recirculation flowrate when tested pursuant to Specification 4.0.5.

c. At least once per 31 days by verifying that the system piping is full of water from the RWT to the header isolation valves. ~~to the 60 inch level in the containment spray header (≥115 foot level).~~

d. At least once per 18 months, during shutdown, by:
   1. Verifying that each automatic valve in the flow path actuates to its correct position on a containment spray actuation (CSAS) and recirculation actuation (RAS) test signal.
   2. Verifying that upon a recirculation actuation test signal, the containment sump isolation valves open and that a recirculation mode flow path via an OPERABLE shutdown cooling heat exchanger is established.

~~*Only when shutdown cooling is not in operation.~~

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SURVEILLANCE REQUIREMENTS (Continued)

3. Verifying that each spray pump starts automatically on ~~a safety injection actuation (SIAS) and on~~ a containment spray actuation (CSAS) test signal.

e. At least once per 5 years by performing an air or smoke flow test through each spray header and verifying each spray nozzle is unobstructed.

SPRAY ADDITIVE

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CONTAINMENT SYSTEMS

~~IODINE REMOVAL~~ SYSTEM

Spray [redacted] Additive

LIMITING CONDITION FOR OPERATION

3.6.2.2 The ~~iodine removal~~ system shall be OPERABLE with:

a. An spray chemical addition tank containing a level of between 90% and 100% (816 and 896 gallons) of between 33% and 35% by weight N_2H_4 solution, and

b. Two spray chemical addition pumps each capable of adding N_2H_4 solution from the spray chemical addition tank to a containment spray system pump flow.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

Spray Additive

With the ~~iodine removal~~ system inoperable, restore the system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours; restore the iodine removal system to OPERABLE status within the next 48 hours or be in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

Spray Additive

4.6.2.2 The ~~iodine removal~~ System shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

b. At least once per 6 months by:

1. Verifying the contained solution volume in the tank, and

2. Verifying the concentration of the N_2H_4 solution by chemical analysis.

c. By verifying that on recirculation flow, each spray chemical addition pump develops a discharge pressure of 100 psig when tested pursuant to Specification 4.0.5

d. At least once per 18 months, during shutdown, by

1. Verifying that each automatic valve in the flow path actuates to its correct position on a containment spray actuation (CSAS) test signal, and

2. Verifying that each spray chemical addition pump starts automatically on a CSAS test signal.

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SURVEILLANCE REQUIREMENTS (Continued)

e. At least once per 5 years by verifying each solution flow rate from the following drain connections in the ~~iodine removal~~ Spray Additive system:

1. ~~SIA-V253 Pump discharge line~~ (Drain line location) 0.63 ± 0.02 gpm.

2. ~~SIB-V254 Pump discharge line~~ (Drain line location) 0.63 ± 0.02 gpm.

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LIMITING CONDITION FOR OPERATION AND SURVEILLANCE REQUIREMENTS

~~3/4.6.2.3 CONTAINMENT COOLING SYSTEM~~ Containment Cooling System

~~See Applicant's SAR.~~

3/4.6.3 Iodine Cleanup System

See Applicant's SAR

3/4.6.~~3~~4 CONTAINMENT ISOLATION VALVES

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LIMITING CONDITION FOR OPERATION

3.6.~~3~~4 The containment isolation valves specified in Table 3.6-1 shall be OPERABLE with isolation times as shown in Table 3.6-1.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

1. With one or more of the isolation valve(s) specified in Table 3.6-1 inoperable, maintain at least one isolation valve OPERABLE in each affected penetration that is open and either:

    a. Restore the inoperable valve(s) to OPERABLE status within 4 hours, or

    b. Isolate each affected penetration within 4 hours by use of at least one deactivated automatic valve secured in the isolation position,* or

    c. Isolate the affected penetration within 4 hours by use of at least one closed manual valve or blind flange;* or

    d. Be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.3.1 The isolation valves specified in Table 3.6-1 shall be demonstrated OPERABLE prior to returning the valve to service after maintenance, repair, or replacement work is performed on the valve or its associated actuator, control, or power circuit by performance of a cycling test and verification of isolation time.

4.6.3.2 Each isolation valve specified in ~~Sections A, B, and C of~~ Table 3.6-1 shall be demonstrated OPERABLE during the COLD SHUTDOWN or REFUELING MODE at least once per 18 months by:

a. Verifying that on ~~a CIAS, CCAS or SIAS~~ its applicable test signal, each isolation valve actuates to its isolation position. ← Insert

b. Verifying that on a Containment Radiation-High test signal, ~~all containment purge valves actuate to their isolation position.~~ **. (BOP)

*The inoperable isolation valve(s) may be part of a system(s). Isolating the affected penetration(s) may affect the use of the system(s). Consider the technical specification requirements on the affected system(s) and act accordingly.

** See Applicant's SAR for [redacted] containment isolation valves actuated by this signal.

Insert

Valves which may be required to open or close following an accident will be actuated to demonstrate their capability to achieve both positions.

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SURVEILLANCE REQUIREMENTS (Continued)

4.6.3.3 The isolation time of each power operated or automatic valve of ~~Sections A, B and C of~~ Table 3.6-1 shall be determined to be within its limit when tested pursuant to Specification 4.0.5.

~~4.6.3.4 The check valves specified in Section D of Table 3.6-1 shall be demonstrated OPERABLE pursuant to 10 CFR 50, Appendix J, with the exception of those check valves footnoted as "Not Type C Tested."~~

~~4.6.3.5 The isolation valves specified in Sections E, F, and G of Table 3.6-1 shall be demonstrated OPERABLE as required by Specification 4.0.5 and the Surveillance Requirements associated with those Limiting Conditions for Operation pertaining to each valve or system in which it is installed. Valves secured** in their actuated position are considered operable pursuant to this specification.~~

~~4.6.3.6 The manual isolation valves specified in Section H of Table 3.6-1 shall be demonstrated OPERABLE pursuant to Surveillance Requirement 4.6.1.1.a of Specification 3.6.1.1.~~

~~**Locked, sealed, or otherwise prevented from unintentional operation.~~

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TABLE 3.6-1
CONTAINMENT ISOLATION VALVES

VALVE NUMBER	PENETRATION NUMBER	FUNCTION	MAXIMUM ACTUATION TIME (SECONDS)
		A. CONTAINMENT ISOLATION (CIAS)	
RDA-UV 023	9	Containment radwaste sump pump to LRS holdup tank	30
RDB-UV 024	9	Containment radwaste sump pump to LRS holdup tank	5
RDB-UV 407	9	Containment radwaste sump post-accident sampling system	5
SGB-HV 200#	11	Downcomer feedwater chemical injection	1
SGB-HV 201#	12	Downcomer feedwater chemical injection	1
SIA-UV 708#	23	Containment recirc sump to post-accident sampling system	5
HCB-UV 044	25A	Containment air radioactivity monitor (inlet)	12
HCA-UV 045	25A	Containment air radioactivity monitor (inlet)	12
HCA-UV 046	25B	Containment air radioactivity monitor (outlet)	12
HCA-UV 047	25B	Containment air radioactivity monitor (outlet)	12
GAA-UV 002	29	N_2 to steam generator and reactor drain tank	10
GAA-UV 001	30	N_2 to SI tanks	10

INSERT

~~TABLE 3.6-1~~

~~CONTAINMENT ISOLATION VALVE ACTUATION TIMES~~

Penetration Number	Valve Number	FUNCTION ~~Description~~	Valve Location Relative to Containment	ESF Actuation Signal	Required Post-Accident Valve Position	Maximum Valve Actuation Time (sec)
A. Remotely Actuated Valves						
11	SI-331	Hot Leg Injection Valve	Outside	None	Open	10
12	SI-321	Hot Leg Injection Valve	Outside	None	Open	10
13 14 15 16	SI-616, 617 SI-626, 627 SI-636, 637 SI-646, 647	High Pressure Cold Leg Injection Valves	Outside	SIAS	Open	10
17 18 19 20	SI-615 SI-625 SI-635 SI-645	Low Pressure Cold Leg Injection Valves	Outside	SIAS	Open	10
23	SI-673	Containment sump isolation valve	Inside	RAS	Open	20
	SI-674	Containment sump isolation valve	Outside	RAS	Open	20
24	SI-675	Containment sump isolation valve	Inside	RAS	Open	20
	SI-676	Containment sump isolation valve	Outside	RAS	Open	20
27	SI-690	Shutdown Cooling Warmup bypass valve	Outside	None	Open or Closed	30
	SI-656	Shutdown Cooling isolation valve	Outside	None	Open or Closed	80
	SI-654	Shutdown Cooling isolation valve	Inside	None	Open or Closed	80

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VALVE NUMBER	PENETRATION NUMBER	FUNCTION	MAXIMUM ACTUATION TIME (SECONDS)
		A. CONTAINMENT ISOLATION (CIAS) (Continued)	
HPA-UV 001	35	Containment to hydrogen recombiner	12
HPA-UV 003	35	Containment to hydrogen recombiner	12
HPA-UV 024	35	H_2 control system	5
HPB-UV 002	36	Containment to hydrogen recombiner	12
HPA-UV 005	38	Containment to hydrogen recombiner	12
HPB-UV 004	36	H_2 recombiner return to containment (inlet)	12
HPA-UV 023	38	H_2 control system	5
HPB-UV 006	39	H_2 recombiner return to containment (inlet)	12
CHA-UV 516	40	Letdown line from RC loop 2B to regenerative heat exchanger and letdown heat exchanger	5
CHB-UV 523	40	Letdown line from RC loop 2B to regenerative heat exchanger and letdown heat exchanger	5
CHB-UV 924	40	Letdown line to post-accident sampling system	5
SSB-UV 201	42A	Pressurizer sample surge line	5
SSA-UV 204	42A	Pressurizer sample surge line	5
SSB-UV 202	42B	Pressurizer sample surge line	5
SSA-UV 205	42B	Pressurizer sample surge line	5
SSB-UV 200	42C	Pressurizer sample surge line	5
SSA-UV 203	42C	Pressurizer sample surge line	5

INSERT

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~~TABLE 3.6-1 (Cont'd)~~

~~CONTAINMENT ISOLATION VALVE ACTUATION TIMES~~

INSERT

Penetration Number	Valve Number	FUNCTION ~~Description~~	Valve Location Relative to Containment	ESF Actuation Signal	Required Post-Accident Valve Position	Maximum Valve Actuation Time (sec)
28	SI-691	Shutdown Cooling Warmup bypass valve	Outside	None	Open or Closed	30
	SI-655	Shutdown Cooling isolation valve	Outside	None	Open or Closed	80
	SI-653	Shutdown Cooling isolation valve	Inside	None	Open or Closed	80
29	SI-682	Safety Injection Tank fill and drain isolation valve	Inside	SIAS	Closed	5
40	CH-523 CH-516	CVCS Letdown Line Isolation Valves	Outside Inside	CIAS CIAS/SIAS	Closed Closed	5 5
41	CH-524	CVCS Charging Line Isolation Valves	Outside	None	Open or Closed	5
43	CH-505 CH-506	Reactor Coolant Pump Controlled Bleedoff Containment Isolation Valves	Outside Inside	CIAS CIAS	Closed Closed	5 5
44	CH-560 CH-561	Reactor Drain Tank Suction Isolation Valves	Inside Outside	CIAS CIAS	Closed Closed	5 5
45	CH-580	Reactor Makeup Water Supply Isolation Valve to the RDT	Outside	CIAS	Closed	5
57	CH-255	Seal Injection Containment Isolation Valve	Outside	None	Open or Closed	5

B. Manual Valves

29	SI-463	Safety Injection Tank Fill and Drain Isolation Valve	Outside	None	Closed	Not Applicable

TABLE 3.6-1 (Continued)
CONTAINMENT ISOLATION VALVES

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VALVE NUMBER	PENETRATION NUMBER	FUNCTION	MAXIMUM ACTUATION TIME (SECONDS)
		A. CONTAINMENT ISOLATION (CIAS) (Continued)	
CHA-UV 560	44	Reactor Drain tank to pre-holdup ion exchanger	5
CHB-UV 561	44	Reactor Drain tank to pre-holdup ion exchanger	5
CHA-UV 580	45	Makeup to reactor drain tank	5
CHA-UV 715	45	Makeup to reactor drain tank post-accident sampling system	5
GRA-UV 001	52	RDT vent to WG surge tank	12
GRB-UV 002	52	RDT vent to WG surge tank	10
WCB-UV 63	60	Normal chilled water to containment ACU (inlet)	10
WCB-UV 61	61	Normal chilled water to containment ACU (outlet)	10
WCA-UV 62	61	Normal chilled water to containment ACU (outlet)	10

- Valve exempt from Type C testing

~~TABLE 3.6-1 (Cont'd)~~

~~CONTAINMENT ISOLATION VALVE ACTUATION TIMES~~

Penetration Number	Valve Number	FUNCTION ~~Description~~	Valve Location Relative to Containment	ESF Actuation Signal	Required Post-Accident Valve Position	Maximum Valve Actuation Time (sec)
41	CH-393 CH-854	CVCS Charging Line Isolation Valves	Inside Outside	None None	Closed Closed	Not Applicable Not Applicable
C. Check Valves						
11 12	SI-533 SI-523	Hot Leg Injection Line Isolation Valve	Inside	None	Open	Not Applicable
13 14 15 16	SI-113 SI-123 SI-133 SI-144	High Pressure Cold Leg Injection Line Isolation Valve	Inside	None	Open	Not Applicable
17 18 19 20	SI-?14 SI-124 SI-134 SI-144	Low Pressure Injection Line Isolation Valves	Inside	None	Open	Not Applicable
41	CH-431 CH-433	CVCS Charging Line Isolation Valves	Inside	None	Open or Closed	Not Applicable
45	CH-494	Reactor Makeup Water Supply Isolation Valve to the RDT	Inside	None	Closed	Not Applicable
57	CH-835	Seal Injection Containment Isolation Valve	Inside	None	Open or Closed	Not Applicable

CONTAINMENT SYSTEMS

3/4.6.4 COMBUSTIBLE GAS CONTROL

HYDROGEN MONITORS

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LIMITING CONDITION FOR OPERATION AND SURVEILLANCE REQUIREMENTS

INSERT

~~3.6.4.1 Two independent containment hydrogen monitors shall be OPERABLE.~~

APPLICABILITY: MODES 1 and 2.

ACTION:

a. With one hydrogen monitor inoperable, restore the inoperable monitor to OPERABLE status within 30 days or be in at least HOT STANDBY within the next 6 hours.

b. With both hydrogen monitors inoperable, restore at least one monitor to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours.

c. With one hydrogen monitor inoperable, and the hydrogen monitor in the Post Accident Sampling System OPERABLE, the provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.6.4.1 Each hydrogen monitor shall be demonstrated OPERABLE by the performance of a CHANNEL CHECK at least once per 12 hours, a CHANNEL FUNCTIONAL TEST at least once per 31 days, and at least once per 92 days on a STAGGERED TEST BASIS by performing a CHANNEL CALIBRATION using sample gases containing a nominal:

a. One volume percent hydrogen, balance nitrogen.

b. Four volume percent hydrogen, balance nitrogen.

INSERT

3/4.6.5 Combustible Gas Control

3/4.6.5.1 Hydrogen Monitors

3/4.6.5.2 Electric Hydrogen Recombiners

3/4.6.5.3 Hydrogen Purge Cleanup System (Optional)

3/4.6.5.4 Hydrogen Mixing System

3/4.6.6 Penetration Room Exhaust Air Cleanup System (Optional)

3/4.6.7 Vacuum Relief Valves (Optional)

3/4.6.8 Secondary Containment (Dual Type Containment, Optional)

3/4.6.8.1 Shield Building Air Cleanup System

3/4.6.8.2 Shield Building Integrity

3/4.6.8.3 Shield Building Structural Integrity

3/4.7.1 TURBINE CYCLE

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SAFETY VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.1 All main steam safety valves shall be OPERABLE with lift settings as specified in Table 3.7-1.

APPLICABILITY: MODES 1, 2, 3, and 4*.

Maximum Variable Overpower

ACTION:

a. With both reactor coolant loops and associated steam generators in operation and with one or more** main steam safety valves inoperable per steam generator, operation in MODES 1, 2, and 3 may proceed provided that within 4 hours, either all the inoperable valves are restored to OPERABLE status or the ~~Power Level-High~~ trip setpoint is reduced per Table 3.7-2; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. Operation in MODES 3 and 4* may proceed with one reactor coolant loop and associated steam generator in operation, provided that there are no more than four inoperable main steam safety valves associated with the operating steam generator; otherwise, be in COLD SHUTDOWN within the following 30 hours.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.1.1 No additional Surveillance Requirements other than those required by Specification 4.0.5.

*Until the steam generators are no longer required for heat removal.

**The maximum number of inoperable safety valves on any operating steam generator is four (4).

CESSAR-NSSS-STS ~~PALO VERDE UNIT~~

TABLE 3.7-1

STEAM LINE SAFETY VALVES PER LOOPS

VALVE NUMBER†#		LIFT SETTING (±1%)*	MINIMUM RATED CAPACITY**
S/G No. 1	S/G No. 2		
a. ~~SGE-PSV-572~~	~~SGE-PSV-564~~	125~~0~~5 psig	919,000 ~~941,543~~ lb/hr
b. ~~SGE-PSV-579~~	~~SGE-PSV-561~~	125~~0~~5 psig	904,000 ~~941,543~~ lb/hr
c. ~~SGE-PSV-573~~	~~SGE-PSV-555~~	1290 psig	931,000 ~~971,332~~ lb/hr
d. ~~SGE-PSV-578~~	~~SGE-PSV-560~~	1290 psig	931,000 ~~971,332~~ lb/hr
e. ~~SGE-PSV-574~~	~~SGE-PSV-556~~	1315 psig	[illegible] ~~989,950~~ lb/hr
f. ~~SGE-PSV-575~~	~~SGE-PSV-557~~	1315 psig	950,000 ~~989,950~~ lb/hr
g. ~~SGE-PSV-576~~	~~SGE-PSV-558~~	1315 psig	950,000 ~~989,950~~ lb/hr
h. ~~SGE-PSV-577~~	~~SGE-PSV-559~~	1315 psig	950,000 ~~989,950~~ lb/hr
i. ~~SGE-PSV-691~~	~~SGE-PSV-694~~	1315 psig	950,000 ~~989,950~~ lb/hr
j. ~~SGE-PSV-692~~	~~SGE-PSV-695~~	1315 psig	950,000 ~~988,060~~ lb/hr

*The lift setting pressure shall correspond to ambient conditions at the valve at nominal operating temperature and pressure.

**Capacity is rated at lift setting +3% accumulation.

†See Applicant's SAR

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TABLE 3.7-2

MAXIMUM ALLOWABLE STEADY STATE POWER LEVEL AND MAXIMUM VARIABLE OVERPOWER TRIP SETPOINT WITH INOPERABLE STEAM LINE SAFETY VALVES

MAXIMUM NUMBER OF INOPERABLE SAFETY VALVES ON ANY OPERATING STEAM GENERATOR	MAXIMUM VARIABLE OVERPOWER TRIP SETPOINT (% OF RATED THERMAL POWER)	MAXIMUM ALLOWABLE STEADY STATE POWER LEVEL (% OF RATED THERMAL POWER)
1	~~100.0~~ 103.9	~~90.2~~ 94.1
2	~~87.1~~ 93.5	~~87.9~~ 83.7
3	~~86.0~~ 84.0	~~76.4~~ 73.2
4	~~75.0~~ 72.6	~~65.6~~ 62.8

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AUXILIARY FEEDWATER SYSTEM

LIMITING CONDITION FOR OPERATION AND SURVEILLANCE REQUIREMENTS

~~3.7.1.2 At least three independent steam generator auxiliary feedwater pumps~~ and associated flow paths shall be OPERABLE with:

a. Two feedwater pumps, each capable of being powered from separate OPERABLE emergency busses, and

b. One feedwater pump capable of being powered from an OPERABLE steam supply system.

APPLICABILITY: MODES 1, 2, 3, and 4*.

ACTION:

a. With one auxiliary feedwater pump inoperable, restore the required auxiliary feedwater pumps to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With two auxiliary feedwater pumps inoperable be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

c. With three auxiliary feedwater pumps inoperable, immediately initiate corrective action to restore at least one auxiliary feedwater pump to OPERABLE status as soon as possible.

INSERT

SURVEILLANCE REQUIREMENTS

4.7.1.2 Each auxiliary feedwater pump shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by:

1. Testing the turbine-driven pump and both motor-driven pumps pursuant to Specification 4.0.5. The provisions of Specification 4.0.4 are not applicable for the turbine-driven pump for entry into MODE 3.

2. Verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

3. Verifying that all manual valves in the suction lines from the primary AFW supply tank (condensate storage tank CTE-T01) to each AFW pump, and the manual discharge line valve of each AFW pump are locked, sealed or otherwise secured in the open position.

~~*Until the steam generators are no longer required for heat removal~~

Insert

3/4.7.1.2 Auxiliary Feedwater System

See Applicant's SAR

PLANT SYSTEMS

CONDENSATE STORAGE TANK

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LIMITING CONDITION FOR OPERATION

3.7.1.3 The ** condensate storage tank (CST) shall be OPERABLE ~~with a level of at least * feet (300,000 gallons).~~ with a water volume of at least 300,000 gallons

APPLICABILITY: MODES 1, 2, 3# and 4.*#

ACTION:

With the condensate storage tank inoperable, within 4 hours either:

a. Restore the CST to OPERABLE status or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours, or

b. Demonstrate the OPERABILITY of the ~~reactor makeup water tank as a~~ (alternate water source)** as a backup supply to the ~~auxiliary~~ emergency feedwater pumps and restore the condensate storage tank to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN with a OPERABLE shutdown cooling loop in operation within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.3.1 The condensate storage tank shall be demonstrated OPERABLE at least once per 12 hours by verifying the level (contained water volume) is within its limits when the tank is the supply source for the ~~auxiliary~~ emergency feedwater pumps.

4.7.1.3.2 The ~~reactor makeup water tank~~ (alternate water source)** shall be demonstrated OPERABLE at least once per 12 hours whenever ~~the reactor makeup water tank~~ it is the supply source for the auxiliary feedwater pumps by verifying:

a. That the ~~reactor makeup water tank supply line~~ (alternate water source) to the ~~auxiliary~~ emergency feed system isolation valves ~~is open, and~~ are either open or OPERABLE, and

b. That the ~~reactor makeup water tank~~ (alternate water source) contains a water level of at least ~~* feet~~ (300,000 gallons).

---

*Until the steam generators are no longer required for heat removed.

#Not applicable when cooldown is in progress.

** See Applicant's SAR.

LIMITING CONDITION FOR OPERATION

3.7.1.4 The specific activity of the secondary coolant system shall be less than or equal to 0.10 microcurie/gram DOSE EQUIVALENT I-131.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With the specific activity of the secondary coolant system greater than 0.10 microcurie/gram DOSE EQUIVALENT I-131, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.4 The specific activity of the secondary coolant system shall be determined to be within the limit by performance of the sampling and analysis program of Table 4.7-1.

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TABLE 4.7-1

SECONDARY COOLANT SYSTEM SPECIFIC ACTIVITY
SAMPLE AND ANALYSIS PROGRAM

TYPE OF MEASUREMENT AND ANALYSIS	SAMPLE AND ANALYSIS FREQUENCY
1. Gross Activity Determination	At least once per 72 hours
2. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	(a) 1 per 31 days, whenever the gross activity determination indicates iodine concentrations greater than 10% of the allowable limit.
	(b) 1 per 6 months, whenever the gross activity determination indicates iodine concentrations below 10% of the allowable limit.

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MAIN STEAM LINE ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.5 Each main steam line isolation valve shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

MODE 1:

With one main steam line isolation valve inoperable but open, POWER OPERATION may continue provided the inoperable valve is restored to OPERABLE status within 4 hours; otherwise, be in at least MODE 2 within the next 6 hours.

MODES 2, 3, and 4:

With one main steam line isolation valve inoperable, subsequent operation in MODE 2, 3, or 4 may proceed provided:

a. The isolation valve is maintained closed.

b. The provisions of Specification 3.0.4 are not applicable.

Otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.5.1 Each main steam line isolation valve shall be demonstrated OPERABLE by verifying full closure within 4.6 seconds when tested pursuant to Specification 4.0.5.

4.7.1.5.2 The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 or MODE 4 to perform the surveillance testing of Specification 4.7.1.5.1 provided the testing is performed within 12 hours after achieving normal operating steam pressure and normal operating temperature for the secondary side to perform the test.

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LIMITING CONDITION FOR OPERATION

3.7.1.6 ~~The~~ Each atmospheric dump valve~~s~~ and its associated block valve shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.*

ACTION:

a. With less than ~~one~~ two atmospheric dump valves per steam generator OPERABLE, restore the required atmospheric dump valves to OPERABLE status within 72 hours; or be in at least HOT STANDBY within the next 6 hours and HOT SHUTDOWN within the following 6 hours.

b. [redacted] Insert

SURVEILLANCE REQUIREMENTS

4.7.1.6 Each atmospheric dump valve ~~and its associated block valve~~ shall be demonstrated OPERABLE:

a. At least once per 24 hours by verifying that the ~~nitrogen accumulator tank is at a pressure ≥ 400 PSIG~~ **

b. Prior to startup following any refueling shutdown or cold shutdown of 30 days or longer, ~~verify~~ by verifying that all valves will open and close fully.

** See Applicant's SAR for means to operate the atmospheric dump valves.

*When steam generators are being used for decay heat removal.

Insert

b. With any block valve(s) inoperable, restore the block valve(s) to OPERABLE status within 7 days; or be in HOT STANDBY within 6 hours and HOT SHUTDOWN within the following 6 hours. Limit the use of the associated atmospheric dump valve(s) during this period.

3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

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LIMITING CONDITION FOR OPERATION

3.7.2 The temperature of the secondary coolant in the steam generators shall be greater than ~~110~~ ** °F when the pressure of the secondary coolant in the steam generator is greater than ~~210~~ ** psig.

APPLICABILITY: At all times.

ACTION:

With the requirements of the above specification not satisfied:

a. Reduce the steam generator pressure to less than or equal to ~~210~~ ** psig within 30 minutes, and

b. Perform an engineering evaluation to determine the effect of the overpressurization on the structural integrity of the steam generator. Determine that the steam generator remains acceptable for continued operation prior to increasing its temperatures above ~~210~~ ** °F.

SURVEILLANCE REQUIREMENTS

4.7.2 The pressure in the secondary side of the steam generators shall be determined to be less than ~~210~~ ** psig at least once per ** hours when the temperature of the secondary coolant is less than ~~110~~ ** °F.

** See Applicant's SAR

PLANT SYSTEMS COMPONENT

3/4.7.3 ~~ESSENTIAL~~ COOLING WATER SYSTEM

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LIMITING CONDITION FOR OPERATION AND SURVEILLANCE REQUIREMENTS

~~3.7.3 At least two independent essential cooling water loops shall be~~ OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With only one essential cooling water loop OPERABLE, restore at least two loops to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

INSERT

4.7.3 At least two essential cooling water loops shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) servicing safety-related equipment that is not locked, sealed, or otherwise secured in position, is in its correct position.

b. At least once per 18 months during shutdown, by verifying that each automatic valve servicing safety-related equipment actuates to its correct position on an SIAS test signal.

c. At least once per 18 months during shutdown, by verifying that the essential cooling water pumps start on an SIAS test signal.

d. At least once per 18 months during shutdown, by verifying that each valve (manual, power-operated, or automatic) servicing safety-related equipment that is locked, sealed, or otherwise secured in position, is in its correct position.

Insert

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3/4.7.3 Component Cooling Water

3/4.7.4 Service Water System

3/4.7.5 Ultimate Heat Sink

-3/4.7.6 Flood Protection

3/4.7.7 Control Room Emergency Air Cleanup System

3/4.7.8 ECCS Pump Room Exhaust Air Cleanup System

See Applicant's SAR

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3/4.7.9 SNUBBERS

LIMITING CONDITION FOR OPERATION

3.7.9 All hydraulic and mechanical snubbers shall be OPERABLE. The only snubbers excluded from this requirement are those installed on nonsafety-related systems and then only if their failure or failure of the system on which they are installed, would have no adverse effect on any safety-related system.

APPLICABILITY: MODES 1, 2, 3, and 4. MODES 5 and 6 for snubbers located on systems required OPERABLE in those MODES.

ACTION:

With one or more snubbers inoperable on any system, within 72 hours replace or restore the inoperable snubber(s) to OPERABLE status and perform an engineering evaluation per Specification 4.7.9g. on the attached component or declare the attached system inoperable and follow the appropriate ACTION statement for that system.

SURVEILLANCE REQUIREMENTS

4.7.9 Each snubber shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program and the requirements of Specification 4.0.5.

a. Inspection Types

As used in this specification, type of snubber shall mean snubbers of the same design and manufacturer, irrespective of capacity.

b. Visual Inspections

Snubbers are categorized as inaccessible or accessible during reactor operation. Each of these groups (inaccessible and accessible) may be inspected independently according to the schedule below. The first inservice visual inspection of each type of snubber shall be performed after 4 months but within 10 months of commencing POWER OPERATION and shall include all hydraulic and mechanical snubbers. If all snubbers of each type are found OPERABLE during the first inservice visual inspection, the second inservice visual inspection of that type shall be performed at the first refueling outage. Otherwise, subsequent visual inspections of a given type shall be performed in accordance with the following schedule:

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No. of Inoperable Snubbers of Each Type per Inspection Period	Subsequent Visual Inspection Period*#
0	18 months ± 25%
1	12 months ± 25%
2	6 months ± 25%
3,4	124 days ± 25%
5,6,7	62 days ± 25%
8 or more	31 days ± 25%

c. Visual Inspection Acceptance Criteria

Visual inspections shall verify that: (1) there are no visible indications of damage or impaired OPERABILITY and (2) attachments to the foundation or supporting structure are secure, and (3) fasteners for attachment of the snubber to the component and to the snubber anchorage are secure. Snubbers which appear inoperable as a result of visual inspections may be determined OPERABLE for the purpose of establishing the next visual inspection interval, provided that: (1) the cause of the rejection is clearly established and remedied for that particular snubber and for other snubbers irrespective of type on that system that may be generically suspectible; and (2) the affected snubber is functionally tested in the as-found condition and determined OPERABLE per Specifications 4.7.9f. When a fluid port of a hydraulic snubber is found to be uncovered, the snubber shall be declared inoperable and cannot be determined OPERABLE via functional testing unless the test is started with the piston in the as-found setting, extending the piston rod in the tension mode direction. All snubbers connected to an inoperable common hydraulic fluid reservoir shall be counted as inoperable snubbers. Snubbers which appear inoperable during an area post maintenance inspection, area walkdown, or Transient Event Inspection shall not be considered inoperable for the purpose of establishing the Subsequent Visual Inspection Period provided that the cause of the inoperability is clearly established and remedied for that particular snubber and for the other snubbers, irrespective of type, that may be generally susceptible.

d. Transient Event Inspection

An inspection shall be performed of all hydraulic and mechanical snubbers attached to sections of systems that have experienced unexpected, potentially damaging transients as determined from a review of operational data and a visual inspection of the systems within 6 months following such an event. In addition to satisfying

*The inspection interval for each type of snubber on a given system shall not be lengthened more than one step at a time unless a generic problem has been identified and corrected; in that event the inspection interval may be lengthened one step the first time and two steps thereafter if no inoperable snubbers of that type are found on that system.

#The provisions of Specification 4.0.2 are not applicable.

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the visual inspection acceptance criteria, freedom-of-motion of mechanical snubbers shall be verified using at least one of the following: (1) manually induced snubber movement; or (2) evaluation of in-place snubber piston setting; or (3) stroking the mechanical snubber through its full range of travel.

e. Functional Tests

During the first refueling shutdown and at least once per 18 months thereafter during shutdown, a representative sample of snubbers shall be tested using one of the following sample plans. The sample plan shall be selected prior to the test period and cannot be changed during the test period. The NRC Regional Administrator shall be notified in writing of the sample plan selected prior to the test period or the sample plan used in the prior test period shall be implemented:

1) At least 10% of the total of each type of snubber shall be functionally tested either in-place or in a bench test. For each snubber of a type that does not meet the functional test acceptance criteria of Specification 4.7.9f., an additional 10% of that type of snubber shall be functionally tested until no more failures are found or until all snubbers of that type have been functionally tested; or

2) A representative sample of each type of snubber shall be functionally tested in accordance with Figure 4.7-1. "C" is the total number of snubbers of a type found not meeting the acceptance requirements of Specification 4.7.9f. The cumulative number of snubbers of a type tested is denoted by "N". At the end of each day's testing, the new values of "N" and "C" (previous day's total plus current day's increments) shall be plotted on Figure 4.7-1. If at any time the point plotted falls in the "Reject" region all snubbers of that type shall be functionally tested. If at any time the point plotted falls in the "Accept" region, testing of snubbers of that type may be terminated. When the point plotted lies in the "Continue Testing" region, additional snubbers of that type shall be tested until the point falls in the "Accept" region or the "Reject" region, or all the snubbers of that type have been tested. Testing equipment failure during functional testing may invalidate that day's testing and allow that day's testing to resume anew at a later time, providing all snubbers tested with the failed equipment during the day of equipment failure are retested; or

3) An initial representative sample of 55 snubbers shall be functionally tested. For each snubber type which does not meet the functional test acceptance criteria, another sample of at least one-half the size of the initial sample shall be tested until the total number tested is equal to the initial sample size multiplied by the factor, $1 + C/2$, where "C" is the number of snubbers found which do not meet the functional test acceptance criteria. The results from this sample plan shall be plotted using an "Accept" line which follows the equation $N = 55(1 + C/2)$. Each snubber point should be plotted as soon

as the snubber is tested. If the point plotted falls on or below the "Accept" line, testing of that type of snubber may be terminated. If the point plotted falls above the "Accept" line, testing must continue until the point falls in the "Accept" region or all the snubbers of that type have been tested.

The representative sample selected for the functional test sample plans shall be randomly selected from the snubbers of each type and reviewed before beginning the testing. The review shall ensure as far as practical that they are representative of the various configurations, operating environments, range of size, and capacity of snubbers of each type. Snubbers placed in the same locations as snubbers which failed the previous functional test shall be retested at the time of the next functional test but shall not be included in the sample plan. If during the functional testing, additional sampling is required due to failure of only one type of snubber, the functional testing results shall be reviewed at the time to determine if additional samples should be limited to the type of snubber which has failed the functional testing.

f. Functional Test Acceptance Criteria

The snubber functional test shall verify that:

1) Activation (restraining action) is achieved within the specified range in both tension and compression;

2) Snubber bleed, or release rate where required, is present in both tension and compression, within the specified range;

3) For mechanical snubbers, the force required to initiate or maintain motion of the snubber is within the specified range in both directions of travel; and

4) For snubbers specifically required not to displace under continuous load, the ability of the snubber to withstand load without displacement.

Testing methods may be used to measure parameters indirectly or parameters other than those specified if those results can be correlated to the specified parameters through established methods.

g. Functional Test Failure Analysis

An engineering evaluation shall be made of each failure to meet the functional test acceptance criteria to determine the cause of the failure. The results of this evaluation shall be used, if applicable, in selecting snubbers to be tested in an effort to determine the OPERABILITY of other snubbers irrespective of type which may be subject to the same failure mode.

For the snubbers found inoperable, an engineering evaluation shall be performed on the components to which the inoperable snubbers are attached. The purpose of this engineering evaluation shall be to determine if the components to which the inoperable snubbers are attached were adversely affected by the inoperability of the snubbers in order to ensure that the component remains capable of meeting the designed service.

If any snubber selected for functional testing either fails to lock up or fails to move, i.e., frozen-in-place, the cause will be evaluated and if caused by manufacturer or design deficiency all snubbers of the same type subject to the same defect shall be functionally tested. This testing requirement shall be independent of the requirements stated in Specification 4.7.9e. for snubbers not meeting the functional test acceptance criteria.

h. Functional Testing of Repaired and Replaced Snubbers

Snubbers which fail the visual inspection or the functional test acceptance criteria shall be repaired or replaced. Replacement snubbers and snubbers which have repairs which might affect the functional test result shall be tested to meet the functional test criteria before installation in the unit. These snubbers shall have met the acceptance criteria subsequent to their most recent service, and the functional test must have been performed within 12 months before being installed in the unit.

i. Snubber Seal Replacement Program

The service life of hydraulic and mechanical snubbers shall be monitored to ensure that the service life is not exceeded between surveillance inspections. The maximum expected service life for various seals, springs, and other critical parts shall be determined and established based on engineering information and shall be extended or shortened based on monitored test results and failure history. Critical parts shall be replaced so that the maximum service life will not be exceeded during a period when the snubber is required to be OPERABLE. The parts replacements shall be documented and the documentation shall be retained in accordance with Specification 6.10.2.

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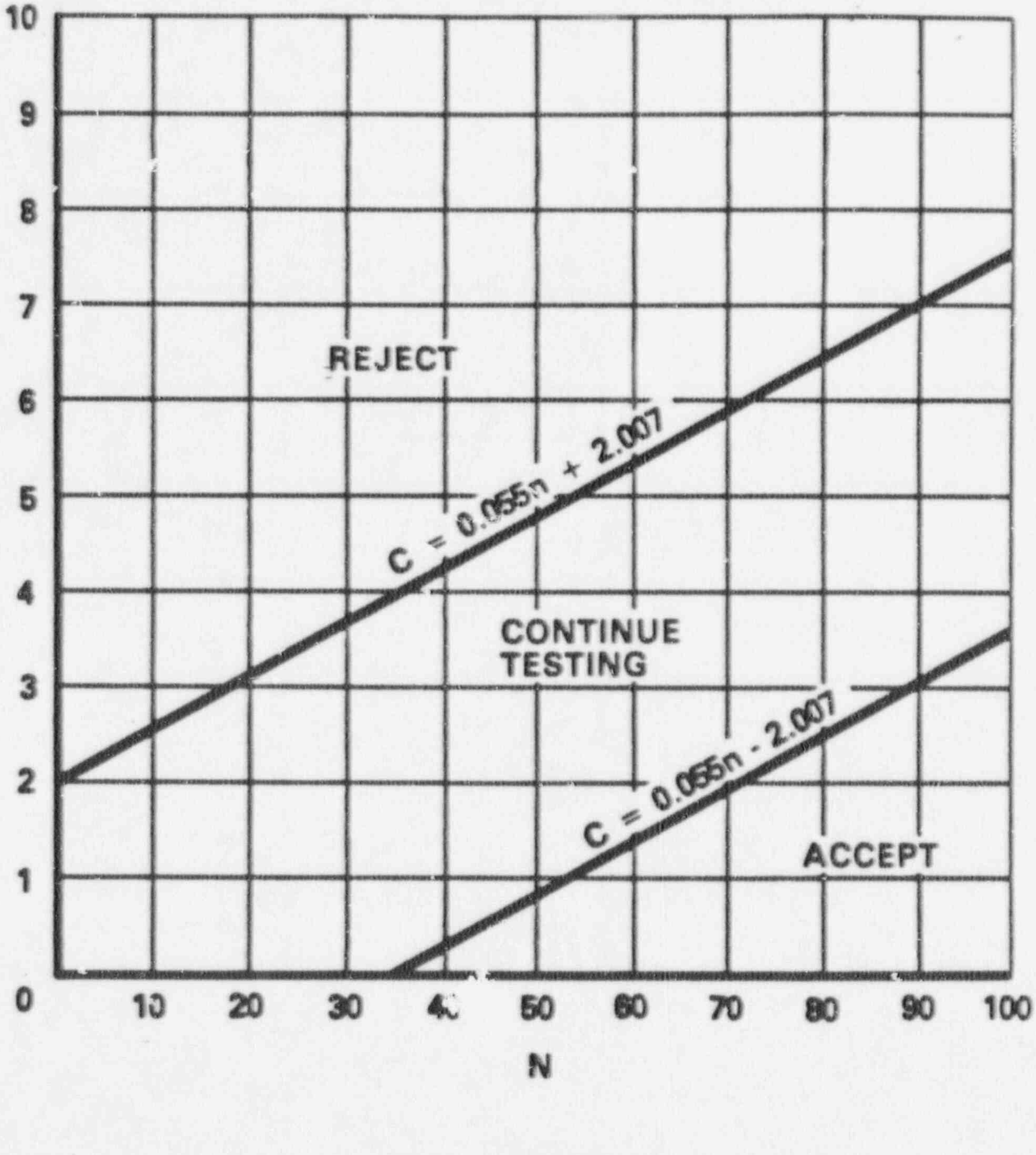

FIGURE 4.7-1

SAMPLING PLAN FOR SNUBBER FUNCTIONAL TEST

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3/4.7.10 SEALED SOURCE CONTAMINATION

LIMITING CONDITION FOR OPERATION AND SURVEILLANCE REQUIREMENTS

3.7.10 Each sealed source containing radioactive material either in excess of 100 microcuries of beta and/or gamma emitting material or 5 microcuries of alpha emitting material shall be free of greater than or equal to 0.005 microcurie of removable contamination.

APPLICABILITY: At all times.

ACTION:

a. With a sealed source having removable contamination in excess of the above limit, immediately withdraw the sealed source from use and either:

1. Decontaminate and repair the sealed source, or
2. Dispose of the sealed source in accordance with Commission Regulations.

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

INSERT

SURVEILLANCE REQUIREMENTS

4.7.10.1 Test Requirements - Each sealed source shall be tested for leakage and/or contamination by:

a. The licensee, or
b. Other persons specifically authorized by the Commission or an Agreement State.

The test method shall have a detection sensitivity of at least 0.005 microcurie per test sample.

4.7.10.2 Test Frequencies - Each category of sealed sources (excluding startup sources and fission detectors previously subjected to core flux) shall be tested at the frequencies described below.

a. Sources in use - At least once per 6 months for all sealed sources containing radioactive material:

1. With a half-life greater than 30 days (excluding Hydrogen 3), and
2. ~~In any form other than gas.~~

Insert

3/4.7.10 Sealed Source Contamination

3/4.7.11 Fire Suppression Systems

3/4.7.11.1 Fire Suppression Water System

3/4.7.11.2 Spray and/or Sprinkler System

Table 3.7-3, Spray and/or Sprinkler System

3/4.7.11.3 CO_2 System

3/4.7.11.4 Fire Hose Stations

Table 3.7-4, Fire Hose Stations

3/4.7.11.5 Yard Fire Hydrants and Hydrant Hose Houses

Table 3.7-5, Yard Fire Hydrants and Associated Hydrant Hose Houses

3/4.7.11.6 Halon Systems

3/4.7.12 Fire-Rated Assemblies

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3/4.7.13 SHUTDOWN COOLING SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.13 Two independent shutdown cooling subsystems shall be OPERABLE, with each subsystem comprised of:

a. One OPERABLE low pressure safety injection pump, and

b. An independent OPERABLE flow path capable of taking suction from the RCS hot leg and discharging coolant through the shutdown cooling heat exchanger and back to the RCS through the cold leg injection lines.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With one shutdown cooling subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 72 hours or be in at least HOT STANDBY within 1 hour, be in at least HOT SHUTDOWN within the next 6 hours and be in COLD SHUTDOWN within the next 30 hours and continue action to restore the required subsystem to OPERABLE status.

b. With both shutdown cooling subsystems inoperable, restore one subsystem to OPERABLE status within 1 hour or be in at least HOT STANDBY within 1 hour and be in HOT SHUTDOWN within the next 6 hours and continue action to restore the required subsystems to OPERABLE status.

c. With both shutdown cooling subsystems inoperable and both reactor coolant loops inoperable, initiate action to restore the required subsystems to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.7.13 Each shutdown cooling subsystem shall be demonstrated OPERABLE:

a. At least once per 18 months, during shutdown, by establishing shutdown cooling flow from the RCS hot legs, through the shutdown cooling heat exchangers, and returning to the RCS cold legs.

b. At least once per 18 months, during shutdown, by testing the automatic and interlock action of the shutdown cooling system connections from the RCS. The shutdown cooling system suction valves shall not open when RCS pressure is greater than ~~[illegible]~~ psia. The shutdown cooling system suction valves located outside containment shall close automatically when RCS pressure is greater than ~~[illegible]~~ psia. The shutdown cooling system suction valve located inside containment shall close automatically when RCS pressure is greater than 700 psia.

3/4.8 ELECTRICAL POWER SYSTEMS

3/4.8.1 A.C. SOURCES

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OPERATING

LIMITING CONDITION FOR OPERATION AND SURVEILLANCE REQUIREMENTS

~~3.8.1.1 As a minimum, the following A.C. electrical power sources shall be~~ OPERABLE:

a. Two physically independent circuits from the offsite transmission network to the switchyard and two physically independent circuits from the switchyard to the onsite Class 1E distribution system, and

b. Two separate and independent diesel generators, each with:

1. Separate day fuel tank with a minimum level of 2.75 feet (550 gallons of fuel), and

2. A separate fuel storage system with a minimum level of 80% (71,500 gallons of fuel), and

3. A separate fuel transfer pump.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With either an offsite circuit or diesel generator of the above required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirements 4.8.1.1.1a. and 4.8.1.1.2a.4 within 1 hour and at least once per 8 hours thereafter; restore at least two offsite circuits and two diesel generators to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one offsite circuit and one diesel generator of the above required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirements 4.8.1.1.1a. and 4.8.1.1.2a.4. within 1 hour and at least once per 8 hours thereafter; restore at least one of the inoperable sources to OPERABLE status within 12 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore at least two offsite circuits and two diesel generators to OPERABLE status within 72 hours from the time of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

INSERT 1

c. With one diesel generator inoperable in addition to ACTION a. or b. above, verify that:

1. All required systems, subsystems, trains, components, and devices that depend on the remaining OPERABLE diesel generator as a source of emergency power are also OPERABLE, and

2. When in MODE 1, 2, 3, or 4*, the steam-driven auxiliary feed pump is OPERABLE.

If these conditions are not satisfied within 2 hours, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

~~*Until the steam generator is no longer required for heat removal.~~

Insert

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3/4.8.1 A.C. Sources

3/4.8.1.1 A.C. Sources Operating

Table 4.8-1, Diesel Generator Test Schedule

3/4.8.1.2 A.C. Sources Shutdown

3/4.8.2 D.C. Sources

3/4.8.2.1 D.C. Sources Operating

Table 3.8-1, D.C. Electrical Sources

Table 4.8-2, Battery Surveillance Requirements

3/4.8.2.2 D.C. Sources Shutdown

3/4.8.3 Onsite Power Distribution Systems

3/4.8.3.1 Onsite Power Distribution System Operating

3/4.8.3.2 Onsite Power Distribution System Operating

3/4.8.4. Electric Equipment Protective Devices

3/4.8.4.1 Containment Penetration Conductor Overcurrent Protective Devices

Table 3.8-2, Containment Penetration Conductor Overcurrent Protective Devices

3/4.8.4.2 Motor-Operated Valves Thermal Overload Protection And Bypass Devices

Table 3.8-3, Motor-Operated Valves Thermal Overload Protection And/Or Bypass Devices

See Applicant's SAR

3/4.9.1 BORON CONCENTRATION

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LIMITING CONDITION FOR OPERATION

3.9.1 With the reactor vessel head closure bolts less than fully tensioned or with the head removed, the boron concentration of all filled portions of the Reactor Coolant System and the refueling ~~[illegible]~~ pool shall be maintained uniform and sufficient to ensure that the more restrictive of the following reactivity conditions is met:

a. Either a K_{eff} of 0.95 or less, or

b. A boron concentration of greater than or equal to 2150 ppm.

APPLICABILITY: MODE 6*.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes and initiate and continue boration at greater than or equal to 40 gpm of a solution containing $\geq$ 4000 ppm boron or its equivalent until K_{eff} is reduced to less than or equal to 0.95 or the boron concentration is restored to greater than or equal to 2150 ppm, whichever is the more restrictive.

SURVEILLANCE REQUIREMENTS

4.9.1.1 The more restrictive of the above two reactivity conditions shall be determined prior to:

a. Removing or unbolting the reactor vessel head, and

b. Withdrawal of any full-length CEA in excess of 3 feet from its fully inserted position within the reactor pressure vessel.

4.9.1.2 The boron concentration of the Reactor Coolant System and the refueling ~~[illegible]~~ pool shall be determined by chemical analysis at least once per 72 hours.

*The reactor shall be maintained in MODE 6 whenever fuel is in the reactor vessel with the reactor vessel head closure bolts less than fully tensioned or with the head removed.

3/4.9.2 INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.9.2 As a minimum, two source range neutron flux monitors shall be OPERABLE and operating, each with continuous visual indication in the control room and one with audible indication in the containment and control room.

APPLICABILITY: MODE 6.

ACTION:

a. With one of the above required monitors inoperable or not operating, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

b. With both of the above required monitors inoperable or not operating, determine the boron concentration of the Reactor Coolant System at least once per 12 hours.

SURVEILLANCE REQUIREMENTS

4.9.2 Each source range neutron flux monitor shall be demonstrated OPERABLE by performance of:

a. A CHANNEL CHECK at least once per 12 hours,

b. A CHANNEL FUNCTIONAL TEST within 8 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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3/4.9.3 DECAY TIME

LIMITING CONDITION FOR OPERATION

3.9.3 The reactor shall be subcritical for at least ~~100~~ 72 hours.

APPLICABILITY: During movement of irradiated fuel in the reactor pressure vessel.

ACTION:

With the reactor subcritical for less than ~~100~~ 72 hours, suspend all operations involving movement of irradiated fuel in the reactor pressure vessel.

SURVEILLANCE REQUIREMENTS

4.9.3 The reactor shall be determined to have been subcritical for at least ~~100~~ 72 hours by verification of the date and time of subcriticality prior to movement of irradiated fuel in the reactor pressure vessel.

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3/4.9.4 CONTAINMENT BUILDING PENETRATIONS

LIMITING CONDITION FOR OPERATION AND SURVEILLANCE REQUIREMENTS

3.9.4 The containment building penetrations shall be in the following status:

a. The equipment door closed and held in place by a minimum of four bolts,

b. A minimum of one door in each airlock is closed, and

c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere shall be either:

1. Closed by an isolation valve, blind flange, or manual valve, or

2. Be capable of being closed by an OPERABLE automatic containment purge valve.

APPLICABILITY: During CORE ALTERATIONS or movement of irradiated fuel within the containment.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or movement of irradiated fuel in the containment building.

INSERT

SURVEILLANCE REQUIREMENTS

4.9.4 ~~Each of~~ the above required containment building penetrations shall be determined to be either in its closed/isolated condition or capable of being closed by an OPERABLE automatic containment purge valve within 72 hours prior to the start of and at least once per 7 days during CORE ALTERATIONS or movement of irradiated fuel in the containment building by:

a. Verifying the penetrations are in their closed/isolated condition, or

b. Testing the containment purge valves per the applicable portions of Specification 4.8.9.

Insert

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3/4.9.4 Containment Penetrations

3/4.9.5 Communications

3/4.9.6 Manipulator Crane

3/4.9.7 Crane Travel — Spent Fuel Storage Pool Building

See Applicant's SAR

3/4.9.8 SHUTDOWN COOLING AND COOLANT CIRCULATION

HIGH WATER LEVEL

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LIMITING CONDITION FOR OPERATION

3.9.8.1 At least one shutdown cooling loop shall be OPERABLE and in operation.*

APPLICABILITY: MODE 6 when the water level above the top of the reactor pressure vessel flange is greater than or equal to 23 feet.

ACTION:

With no shutdown cooling loop OPERABLE and in operation, suspend all operations involving an increase in the reactor decay heat load or a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required shutdown cooling loop to OPERABLE and operating status as soon as possible. Close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere within 4 hours.

SURVEILLANCE REQUIREMENTS

4.9.8.1 At least one shutdown cooling loop shall be verified to be in operation and circulating reactor coolant at a flow rate of greater than or equal to 4000 gpm at least once per 12 hours.

*The shutdown cooling loop may be removed from operation for up to 1 hour per 8-hour period during the performance of CORE ALTERATIONS in the vicinity of the reactor pressure vessel hot legs ~~or during surveillance testing of SCGS pumps.~~

LOW WATER LEVEL

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LIMITING CONDITION FOR OPERATION

3.9.8.2 Two independent shutdown cooling loops shall be OPERABLE and at least one shutdown cooling loop shall be in operation.*

APPLICABILITY: MODE 6 when the water level above the top of the reactor pressure vessel flange is less than 23 feet.

ACTION:

a. With less than the required shutdown cooling loops OPERABLE, immediately initiate corrective action to return the required loops to OPERABLE status, or to establish greater than or equal to 23 feet of water above the reactor pressure vessel flange, as soon as possible.

b. With no shutdown cooling loop in operation, suspend all operations involving an increase in the reactor decay heat load or a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required shutdown cooling loop to operation. Close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere within 4 hours.

SURVEILLANCE REQUIREMENTS

4.9.8.2 At least one shutdown cooling loop shall be verified to be in operation and circulating reactor coolant at a flow rate of greater than or equal to 4000 gpm at least once per 12 hours.

---

*The shutdown cooling loop may be removed from operation for up to 1 hour per 8-hour period during the performance of CORE ALTERATIONS in the vicinity of the reactor pressure vessel hot legs ~~or during surveillance testing of ECCS pumps.~~

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3/4.9.9 CONTAINMENT PURGE VALVE ISOLATION SYSTEM

LIMITING CONDITION FOR OPERATION AND SURVEILLANCE REQUIREMENTS

~~3.9.9 The containment purge valve isolation system shall be OPERABLE.~~

APPLICABILITY: During CORE ALTERATIONS or movement of irradiated fuel within the containment.

ACTION:

With the containment purge valve isolation system inoperable, close each of the containment purge penetrations providing direct access from the containment atmosphere to the outside atmosphere. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.9 The containment purge valve isolation system shall be demonstrated OPERABLE within 72 hours prior to the start of and at least once per 7 days during CORE ALTERATIONS by verifying that containment purge valve isolation occurs on manual initiation and on CPIAS.

3/4.9.9 Containment Purge Valve Isolation System

See Applicant's SAR

3/4.9.10 WATER LEVEL - REACTOR VESSEL

FUEL ASSEMBLIES

LIMITING CONDITION FOR OPERATION

3.9.10.1 At least 23 feet of water shall be maintained over the top of the reactor pressure vessel flange.

APPLICABILITY: During movement of fuel assemblies within the reactor pressure vessel when either the fuel assemblies being moved or the fuel assemblies seated within the reactor pressure vessel are irradiated.

ACTION:

With the requirements of the above specification not satisfied, suspend all operations involving movement of fuel assemblies within the pressure vessel.

SURVEILLANCE REQUIREMENTS

4.9.10.1 The water level shall be determined to be at least its minimum required depth within 2 hours prior to the start of and at least once per 24 hours thereafter during movement of fuel assemblies.

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LIMITING CONDITION FOR OPERATION

3.9.10.2 At least 23 feet of water shall be maintained over the top of the fuel seated in the reactor pressure vessel.

APPLICABILITY: During movement of CEAs within the reactor pressure vessel, when the fuel assemblies seated within the reactor pressure vessel are irradiated.

ACTION:

With the requirements of the above specification not satisfied, suspend all operations involving movement of CEAs within the pressure vessel.

SURVEILLANCE REQUIREMENTS

4.9.10.2 The water level shall be determined to be at least its minimum required depth within 2 hours prior to the start of and at least once per 24 hours thereafter during movement of CEAs.

3/4.9.11 WATER LEVEL - STORAGE POOL

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LIMITING CONDITION FOR OPERATION

3.9.11 At least 23 feet of water shall be maintained over the top of irradiated fuel assemblies seated in the storage racks.

APPLICABILITY: Whenever irradiated fuel assemblies are in the storage pool.

ACTION:

With the requirement of the specification not satisfied, suspend all movement of fuel assemblies and crane operations with loads in the fuel storage areas and restore the water level to within its limit within 4 hours.

SURVEILLANCE REQUIREMENTS

4.9.11 The water level in the storage pool shall be determined to be at least its minimum required depth at least once per 7 days when irradiated fuel assemblies are in the fuel storage pool.

STORAGE POOL AIR CLEANUP SYSTEM

3/4.9.12 ~~FUEL BUILDING ESSENTIAL VENTILATION SYSTEM~~

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LIMITING CONDITION FOR OPERATION

~~3.9.12 Two independent fuel building essential ventilation systems shall be~~ OPERABLE.

APPLICABILITY: Whenever irradiated fuel is in the storage pool.

ACTION:

a. With one fuel building essential ventilation system inoperable, fuel movement within the storage pool or crane operation with loads over the storage pool may proceed provided the OPERABLE fuel building essential ventilation system is capable of being powered from an OPERABLE emergency power source. Restore the inoperable fuel building essential ventilation system to OPERABLE status within 7 days or suspend all operations involving movement of fuel within the storage pool or operation of the fuel handling machine over the storage pool.

b. With no fuel building essential ventilation system OPERABLE, suspend all operations involving movement of fuel within the storage pool or crane operation with loads over the storage pool until at least one fuel building essential ventilation system is restored to OPERABLE status.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.12 The above required fuel building essential ventilation systems shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 15 minutes.

b. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire, or chemical release in any ventilation zone ~~communicating with the system by:~~

(3/4.9.12 Storage Pool Air Cleanup System
See Applicant's SAR

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3/4.10.1 SHUTDOWN MARGIN

LIMITING CONDITION FOR OPERATION

3.10.1 The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 may be suspended for measurement of CEA worth and shutdown margin provided reactivity equivalent to at least the highest estimated CEA worth is available for trip insertion from OPERABLE CEA(s), or the reactor is subcritical by at least the reactivity equivalent of the highest CEA worth.

APPLICABILITY: MODES 2 and 3* ~~and 4*.~~

ACTION:

a. With any full-length CEA not fully inserted and with less than the above reactivity equivalent available for trip insertion, immediately initiate and continue boration at greater than or equal to 40 gpm of a solution containing greater than or equal to 4000 ppm boron or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

b. With all full-length CEAs fully inserted and the reactor subcritical by less than the above reactivity equivalent, immediately initiate and continue boration at greater than or equal to 40 gpm of a solution containing greater than or equal to 4000 ppm boron or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

SURVEILLANCE REQUIREMENTS

4.10.1.1 The position of each full-length and part-length CEA required either partially or fully withdrawn shall be determined at least once per 2 hours.

4.10.1.2 Each CEA not fully inserted shall be demonstrated capable of full insertion when tripped from at least the 50% withdrawn position within 24 hours prior to reducing the SHUTDOWN MARGIN to less than the limits of Specification 3.1.1.1.

4.10.1.3 When in MODE 3 ~~[illegible]~~, the reactor shall be determined to be subcritical by at least the reactivity equivalent of the highest estimated CEA worth or the reactivity equivalent of the highest estimated CEA worth is available for trip insertion from OPERABLE CEAs at least once per 2 hours by consideration of at least the following factors:

a. Reactor Coolant System boron concentration,

b. CEA position,

c. Reactor Coolant System average temperature,

d. Fuel burnup based on gross thermal energy generation,

e. Xenon concentration, and

f. Samarium concentration.

*Operation in MODE 3 ~~[illegible]~~ shall be limited to 6 consecutive hours.
~~[illegible]~~

3/4.10.2 MODERATOR TEMPERATURE COEFFICIENT, GROUP HEIGHT, INSERTION, AND POWER DISTRIBUTION LIMITS

LIMITING CONDITION FOR OPERATION

3.10.2 The moderator temperature coefficient, group height, insertion, and power distribution limits of Specifications 3.1.1.3, 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.2, 3.2.3, 3.2.7, and the Minimum Channels OPERABLE requirement of I.C.1 (CEA Calculators) of Table 3.3-1 may be suspended during the performance of PHYSICS TESTS provided:

a. The THERMAL POWER is restricted to the test power plateau which shall not exceed 85% of RATED THERMAL POWER, and

b. The limits of Specification 3.2.1 are maintained and determined as specified in Specification 4.10.2.2 below.

APPLICABILITY: MODES 1 and 2.

ACTION:

With any of the limits of Specification 3.2.1 being exceeded while the requirements of Specifications 3.1.1.3, 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.2, 3.2.3, 3.2.7, and the Minimum Channels OPERABLE requirement of I.C.1 (CEA Calculators) of Table 3.3-1 are suspended, either:

a. Reduce THERMAL POWER sufficiently to satisfy the requirements of Specification 3.2.1, or

b. Be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.10.2.1 The THERMAL POWER shall be determined at least once per hour during PHYSICS TESTS in which the requirements of Specifications 3.1.1.3, 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.2, 3.2.3, 3.2.7, or the Minimum Channels OPERABLE requirement of I.C.1 (CEA Calculators) of Table 3.3-1 are suspended and shall be verified to be within the test power plateau.

4.10.2.2 The linear heat rate shall be determined to be within the limits of Specification 3.2.1 by monitoring it continuously with the Incore Detector Monitoring System pursuant to the requirements of Specifications 4.2.1.3 and 3.3.3.2 during PHYSICS TESTS above 20% of RATED THERMAL POWER in which the requirements of Specifications 3.1.1.3, 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.2, 3.2.3, 3.2.7, or the Minimum Channels OPERABLE requirement of I.C.1 (CEA Calculators) of Table 3.3-1 are suspended.

3/4.10.3 REACTOR COOLANT LOOPS

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LIMITING CONDITION FOR OPERATION

3.10.3 The limitations of Specification 3.4.1.1 and noted requirements of Tables 2.2-1 and 3.3-1 may be suspended during the performance of startup PHYSICS TESTS, provided:

a. The THERMAL POWER does not exceed 5% of RATED THERMAL POWER, and

b. The reactor trip setpoints of the OPERABLE power level channels are set at less than or equal to 20% of RATED THERMAL POWER.

c. Both reactor coolant loops and at least one reactor coolant pump in each loop are in operation.

APPLICABILITY: During startup PHYSICS TESTS.

ACTION:

With the THERMAL POWER greater than 5% of RATED THERMAL POWER or with less than the above required reactor coolant loops in operation and circulating reactor coolant, immediately trip the reactor.

SURVEILLANCE REQUIREMENTS

4.10.3.1 The THERMAL POWER shall be determined to be less than or equal to 5% of RATED THERMAL POWER at least once per hour during startup PHYSICS TESTS.

4.10.3.2 Each logarithmic and variable overpower level neutron flux monitoring channel shall be subjected to a CHANNEL FUNCTIONAL TEST within 12 hours prior to initiating startup PHYSICS TESTS.

4.10.3.3 The above required reactor coolant loops shall be verified to be in operation and circulating reactor coolant at least once per 12 hours.

3/4.10.4 CEA POSITION, REGULATING CEA INSERTION LIMITS AND REACTOR COOLANT COLD LEG TEMPERATURE

LIMITING CONDITION FOR OPERATION

3.10.4 The requirements of Specifications 3.1.3.1, 3.1.3.6 and 3.2.6 may be suspended during the performance of PHYSICS TESTS to determine the isothermal temperature coefficient, moderator temperature coefficient, and power coefficient provided: ~~the limits of Specification 3.2.1 are maintained and determined as specified in Specification 4.10.4.2 below.~~

INSERT 1

APPLICABILITY: MODES 1 and 2.

ACTION:

With any of the limits of Specification 3.2.1 being exceeded while the requirements of Specifications 3.1.3.1, 3.1.3.6 and 3.2.6 are suspended, either:

a. Reduce THERMAL POWER sufficiently to satisfy the requirements of Specification 3.2.1, or

b. Be in HOT STANDBY within 6 hours.

Insert 2

SURVEILLANCE REQUIREMENTS

4.10.4.1 The THERMAL POWER shall be determined at least once per hour during PHYSICS TESTS in which the requirements of Specifications 3.1.3.1, 3.1.3.6 and/or 3.2.6 are suspended and shall be verified to be within the test power plateau.

4.10.4.2 The linear heat rate shall be determined to be within the limits of Specification 3.2.1 by monitoring it continuously with the Incore Detector Monitoring System pursuant to the requirements of Specification 3.3.3.2 during PHYSICS TESTS above 20% of RATED THERMAL POWER in which the requirements of Specifications 3.1.3.1, 3.1.3.6 and/or 3.2.6 are suspended.

Insert 3

INSERT 1

a. THE LIMITS OF SPECIFICATION 3.2.1 ARE MAINTAINED AND DETERMINED AS SPECIFIED IN SPECIFICATION 4.10.4.2 BELOW.

b. THE CONDENSER VACUUM IS MAINTAINED GREATER THAN * INCHES OF MERCURY AND DETERMINED AS SPECIFIED IN SPECIFICATION 4.10.4.3 BELOW.

c. THE REACTOR COOLANT COLD LEG TEMPERATURE BY NARROW RANGE INDICATION DOES NOT EXCEED 568 °F, DETERMINED AS SPECIFIED IN SPECIFICATION 4.10.4.4 BELOW.

d. THE REACTOR COOLANT COLD LEG TEMPERATURE BY NARROW RANGE INDICATION IS NOT LESS THAN 552 °F, DETERMINED AS SPECIFIED IN SPECIFICATION 4.10.4.4 BELOW.

Insert 2

WITH THE CONDENSER VACUUM LESS THAN OR EQUAL TO * INCHES OF MERCURY WHILE THE REQUIREMENTS OF SPECIFICATION 3.2.6 ARE SUSPENDED, EITHER:

1. IMMEDIATELY REDUCE THERMAL POWER TO SATISFY THE REQUIREMENTS OF SPECIFICATION 3.2.6, OR
2. TRIP THE REACTOR.

WITH THE REACTOR COOLANT COLD LEG TEMPERATURE GREATER THAN 568 °F OR LESS THAN 552 °F WHILE THE REQUIREMENTS OF SPECIFICATION 3.2.6 ARE SUSPENDED:

1. RESTORE THE TEMPERATURE TO WITHIN THE LIMITS WITHIN 2 HOURS, OR

2. REDUCE THE THERMAL POWER TO LESS THAN 30% OF RATED THERMAL POWER WITHIN THE NEXT 4 HOURS.

Insert 3

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4.10.4.3 THE CONDENSER VACUUM SHALL BE DETERMINED TO BE GREATER THAN * INCHES OF MERCURY BY MONITORING IT AT LEAST ONCE PER 30 MINUTES DURING PHYSICS TESTS IN WHICH THE REQUIREMENTS OF SPECIFICATION 3.2.6 ARE SUSPENDED

4.10.4.4 THE REACTOR COOLANT COLD LEG TEMPERATURE SHALL BE DETERMINED TO BE WITHIN THE LIMITS OF SPECIFICATIONS 3.10.4.c AND 3.10.4.d BY MONITORING THE NARROW RANGE TEMPERATURE INDICATION AT LEAST ONCE PER HOUR DURING PHYSICS TESTS IN WHICH THE REQUIREMENTS OF SPECIFICATION 3.2.6 ARE SUSPENDED

* - SEE APPLICANT'S SAR

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3/4.10.5 MINIMUM TEMPERATURE AND PRESSURE FOR CRITICALITY

LIMITING CONDITION FOR OPERATION

3.10.5 The minimum temperature and pressure for criticality limits of Specifications 3.1.1.4 and 3.2.8 may be suspended during low temperature PHYSICS TESTS to a minimum temperature of 300°F and a minimum pressure of 500 psia provided:

a. The THERMAL POWER does not exceed 5% of RATED THERMAL POWER.

b. The reactor trip setpoints on the OPERABLE Variable Overpower trip channels are set at ≤ 20% of RATED THERMAL POWER, and

c. The Reactor Coolant System temperature and pressure relationship is maintained within the acceptable region of operation required by Specification 3.4.8 except that the core critical line shown on Figure 3.4-2 does not apply.

APPLICABILITY: MODE 2*.

ACTION:

a. With the THERMAL POWER greater than 5% of RATED THERMAL POWER, immediately open the reactor trip breakers.

b. With the Reactor Coolant System temperature and pressure relationship within the region of unacceptable operation on Figure 3.4-2, immediately open the reactor trip breakers and restore the temperature-pressure relationship to within its limit within 30 minutes; perform the engineering evaluation required by Specification 3.4.8.1 prior to the next reactor criticality.

SURVEILLANCE REQUIREMENTS

4.10.5.1 The Reactor Coolant System temperature and pressure relationship shall be verified to be within the acceptable region for operation of Figure 3.4-2 at least once per hour.

4.10.5.2 The THERMAL POWER shall be determined to be ≤ 5% of RATED THERMAL POWER at least once per hour.

4.10.5.3 The Reactor Coolant System temperature shall be verified to be greater than or equal to 300°F at least once per hour.

4.10.5.4 Each Logarithmic Power Level and Variable Overpower channel shall be subjected to a CHANNEL FUNCTIONAL TEST within 12 hours prior to initiating low temperature PHYSICS TESTS.

*First core only, prior to first exceeding 5% RATED THERMAL POWER.

3/4.10.6 SAFETY INJECTION TANKS

LIMITING CONDITION FOR OPERATION

3.10.6 The safety injection tank isolation valve requirement of Specification 3.5.1a. may be suspended during partial stroke testing of the low pressure safety injection check valves (SI-114, SI-124, SI-134, SI-144) provided:

a. That power to the isolation valve is restored and the SIAS signal is not overridden.

b. Only one isolation valve at a time is closed during the testing for no longer than 1 hour.

c. That the valve is key locked opened with power removed before the next isolation valve is closed.

APPLICABILITY:

While partial stroke testing of the low pressure injection check valves during normal plant operation.

ACTION:

If the requirement of Specification 3.5.1a. was suspended to perform the Specification 3.10.6 partial stroke test and if any of the Specification 3.10.6 requirements are not met during the Specification 3.10.6 partial stroke testing, the Limiting Condition for Operation shall revert to Specification 3.5.1 and the 3.5.1 ACTION shall be applicable.

SURVEILLANCE REQUIREMENTS

4.10.6.1 A valve alignment shall be performed within 4 hours following completion of testing to verify that all valves operated during this testing are restored to their normal positions and that power is removed to the SIT isolation valves.

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3/4.10.~~8~~7 SAFETY INJECTION TANK PRESSURE

LIMITING CONDITION FOR OPERATION

3.10.~~8~~7 The safety injection tank (SIT) pressure of Specification 3.5.1d. may be suspended, for low temperature PHYSICS TESTS provided:

a. The THERMAL POWER does not exceed 5% of RATED THERMAL POWER;

b. The SITs have been filled per Specification 3.5.1b. and pressurized to 175 to 225 psig below the RCS pressure ~~or not to go below~~ but not less than 254 psig;

c. All valves in the injection lines from the SITs to the RCS are open and the SITs are capable of injecting into the RCS if there is a decrease in RCS pressure.

APPLICABILITY: MODES 2 ~~and~~, 3 and 4

ACTION:

If all the SITs do not meet the level and pressure requirements of Specification 3.10.~~8~~7, restore all the SITs to meet these requirements or be in HOT STANDBY within 6 hours and be in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.10.~~8~~7.1 The THERMAL POWER shall be determined to be less than 5% of RATED THERMAL POWER at least once per hour during low pressure PHYSICS TESTS.

4.10.~~8~~7.2 Every 8 hours verify:

a. All the SITs levels meet the requirements of Specification 3.5.1b.

b. All the SITs pressures meet the requirements of Specification 3.10.8.

c. The valve alignment from the SITs to the RCS has not changed.

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3/4.11 RADIOACTIVE EFFLUENTS

3/4.11.1 SECONDARY SYSTEM LIQUID WASTE DISCHARGES TO ONSITE EVAPORATION PONDS

CONCENTRATION

LIMITING CONDITION FOR OPERATION AND

3.11.1.1 The concentration of radioactive material discharged from secondary system liquid waste to the onsite evaporation ponds shall be limited to the lower limit of detectability (LLD) defined as 5×10^{-7} µCi/ml for the principal gamma emitters or 1×10^{-6} µCi/ml for I-131.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

When any secondary system liquid waste discharge pathway concentration determined in accordance with the surveillance requirements given below exceeds the specified LLD, divert that discharge pathway to the liquid radwaste system without delay.

SURVEILLANCE REQUIREMENTS

4.11.1.1.1 Radioactive liquid wastes collected in the chemical waste neutralizer tank shall be sampled and analyzed prior to their batchwise discharge to the onsite evaporation pond in accordance with the sampling and analysis program specified in Table 4.11-1.

4.11.1.1.2 With the concentration of radioactive material in the chemical waste neutralizer tank exceeding the specified LLD, sample and analyze other secondary system discharge pathways in accordance with the sampling and analysis program specified in Table 4.11-1.

INSERT

Insert

3/4.11.1 Secondary System Liquid Waste Discharges To [redacted] Onsite Evaporation Ponds

3/4.11.1.1 Concentration

Table 4.11-1, Radioactive Liquid Waste Sampling and Analysis program

3/4.11.1.2 Dose

3/4.11.1.3 Liquid Holdup Tanks

3/4.11.2 Gaseous Effluents

3/4.11.2.1 Dose Rate

Table 4.11-2, Radioactive Gaseous Waste Sampling and Analysis Program

3/4.11.2.2 Dose-Noble Gases

3/4.11.2.3 Dose - Iodine-131, Iodine-133, Tritium, And Radionuclides In Particulate Form

3/4.11.2.4 Gaseous Radwaste Treatment

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3/4.11.2.5 Explosive Gas Mixture

3/4.11.2.6 Gas Storage Tanks

3/4.11.3 Solid Radioactive Waste

3/4.11.4 Total Dose

See Applicant's SAR

3/4.12 RADIOLOGICAL ENVIRONMENTAL MONITORING

3/4.12.1 MONITORING PROGRAM

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LIMITING CONDITION FOR OPERATION AND SURVEILLANCE REQUIREMENTS

INSERT

~~3.12.1 The radiological environmental monitoring program shall be conducted~~ as specified in Table 3.12-1.

APPLICABILITY: At all times.

ACTION:

a. With the radiological environmental monitoring program not being conducted as specified in Table 3.12-1, prepare and submit to the Commission, in the Annual Radiological Environmental Operating Report required by Specification 6.9.1.7, a description of the reasons for not conducting the program as required and the plans for preventing a recurrence.

b. With the level of radioactivity as the result of plant effluents in an environmental sampling medium at a specified location exceeding the reporting levels of Table 3.12-2 when averaged over any calendar quarter, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that identifies the cause(s) for exceeding the limit(s) and defines the corrective actions to be taken to reduce radioactive effluents so that the potential annual dose* to A MEMBER OF THE PUBLIC is less than the calendar year limits of Specifications 3.11.1.2, 3.11.2.2, and 3.11.2.3. When more than one of the radionuclides in Table 3.12-2 are detected in the sampling medium, this report shall be submitted if:

$$\frac{\text{concentration (1)}}{\text{reporting level (1)}} + \frac{\text{concentration (2)}}{\text{reporting level (2)}} + \ldots \geq 1.0$$

When radionuclides other than those in Table 3.12-2 are detected and are the result of plant effluents, this report shall be submitted if the potential annual dose* to A MEMBER OF THE PUBLIC is equal to or greater than the calendar year limits of Specifications 3.11.1.2, 3.11.2.2, and 3.11.2.3. This report is not required if the measured level of radioactivity was not the result of plant effluents; however, in such an event, the condition shall be reported and described in the Annual Radiological Environmental Operating Report.

c. With milk or fresh leafy vegetable samples unavailable from one or more of the sample locations required by Table 3.12-1, identify locations for obtaining replacement samples and add them to the radiological environmental monitoring program within 30 days. The specific

*The methodology and parameters used to estimate the potential annual dose to ~~A MEMBER OF THE PUBLIC shall be indicated in this report.~~

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3/4.12.1 Monitoring Program

Table 3.12-1, Radiological Environmental Monitoring Program

Table 3.12-2, Reporting Levels For Radioactivity Concentrations In Environmental Samples

Table 4.12-1, Detection Capabilities For Environmental Sample Analysis

3/4.12.2 Land Use Census

3/4.12.3 Interlaboratory Comparison Program

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BASES

FOR

SECTIONS 3.0 AND 4.0

LIMITING CONDITIONS FOR OPERATION

AND

SURVEILLANCE REQUIREMENTS

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NOTE

The BASES contained in the succeeding pages summarize the reasons for the specifications of Sections 3.0 and 4.0 but in accordance with 10 CFR 50.36 are not a part of these Technical Specifications.

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BASES

The specifications of this section provide the general requirements applicable to each of the Limiting Conditions for Operation and Surveillance Requirements within Section 3/4.

3.0.1 This specification defines the applicability of each specification in terms of defined OPERATIONAL MODES or other specified conditions and is provided to delineate specifically when each specification is applicable.

3.0.2 This specification defines those conditions necessary to constitute compliance with the terms of an individual Limiting Condition for Operation and associated ACTION requirement.

3.0.3 This specification delineates the measures to be taken for circumstances not directly provided for in the ACTION statements and whose occurrence would violate the intent of a specification. For example, Specification 3.6.2.1 requires two containment spray systems to be OPERABLE and provides explicit ACTION requirements if one spray system is inoperable. Under the terms of Specification 3.0.3, if both of the required containment spray systems are inoperable, within 1 hour measures must be initiated to place the unit in at least HOT STANDBY within the next 6 hours, in at least HOT SHUTDOWN within the following 6 hours, and in COLD SHUTDOWN in the subsequent 24 hours.

3.0.4 This specification provides that entry into an OPERATIONAL MODE or other specified applicability condition must be made with (a) the full complement of required systems, equipment, or components OPERABLE and (b) all other parameters as specified in the Limiting Conditions 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 operation is not initiated with either required equipment or systems inoperable or other specified limits being exceeded.

Exceptions to this specification 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.

4.0.1 This specification provides that surveillance activities necessary to ensure the Limiting Conditions for Operation are met and will be performed during the OPERATIONAL MODES or other conditions for which the Limiting Conditions for Operation are applicable. Provisions for additional surveillance activities to be performed without regard to the applicable OPERATIONAL MODES or other conditions are provided in the individual surveillance requirements. Surveillance requirements for Special Test Exceptions need only be performed when the Special Test Exception is being utilized as an exception to an individual specification.

4.0.2 The provisions of this specification provide allowable tolerances for performing surveillance activities beyond those specified in the nominal surveillance interval. These tolerances are necessary to provide operational flexibility because of scheduling and performance considerations. The phrase "at least" associated with a surveillance frequency does not negate this allowable tolerance value and permits the performance of more frequent surveillance activities.

The tolerance values, taken either individually or consecutively over three test intervals, are sufficiently restrictive to ensure that the reliability associated with the surveillance activity is not significantly degraded beyond that obtained from the nominal specified interval.

4.0.3 The provisions of this specification set forth the criteria for determination of compliance with the OPERABILITY requirements of the Limiting Conditions for Operation. Under these criteria, equipment, systems, or components are assumed to be OPERABLE if the associated surveillance activities have been satisfactorily performed within the specified time interval. Nothing in this provision is to be construed as defining equipment, systems, or components OPERABLE, when such items are found or known to be inoperable although still meeting the surveillance requirements.

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4.0.4 This specification ensures that the surveillance activities associated with a Limiting Condition for Operation have been performed within the specified time interval prior to entry into an OPERATIONAL MODE or other applicable condition. The intent of this provision is to ensure that surveillance activities have been satisfactorily demonstrated on a current basis as required to meet the OPERABILITY requirements of the Limiting Condition for Operation.

Under the terms of this specification, for example, during initial plant startup or following extended plant outages, the applicable surveillance activities must be performed within the stated surveillance interval prior to placing or returning the system or equipment into OPERABLE status.

4.0.5 This specification ensures that inservice inspection of ASME Code Class 1, 2, and 3 components and inservice testing of ASME Code Class 1, 2, and 3 pumps and valves will be performed in accordance with a periodically updated version of Section XI of the ASME Boiler and Pressure Vessel Code and Addenda as required by 10 CFR 50.55a. Relief from any of the above requirements has been provided in writing by the Commission and is not a part of these Technical Specifications.

This specification includes a clarification of the frequencies for performing the inservice inspection and testing activities required by Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda. This clarification is provided to ensure consistency in surveillance intervals thoughout these Technical Specifications and to remove any ambiguities relative to the frequencies for performing the required inservice inspection and testing activities.

Under the terms of this specification, the more restrictive requirements of the Technical Specifications take precedence over the ASME Boiler and Pressure Vessel Code and applicable Addenda. For example, the requirements of Specification 4.0.4 to perform surveillance activities prior to entry into an OPERATIONAL MODE or other specified applicability condition takes precedence over the ASME Boiler and Pressure Vessel Code provision which allows pumps to be tested up to 1 week after return to normal operation. And for example, the Technical Specification definition of OPERABLE does not grant a grace period before a device that is not capable of performing its specified function is declared inoperable and takes precedence over the ASME Boiler and Pressure Vessel Code provision which allows a valve to be incapable of performing its specified function for up to 24 hours before being declared inoperable.

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3/4.1.1 BORATION CONTROL

3/4.1.1.1 and 3/4.1.1.2 SHUTDOWN MARGIN

A sufficient SHUTDOWN MARGIN ensures that (1) the reactor can be made subcritical from all operating conditions, (2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits assuming the insertion of the regulating CEAs are within the limits of Specification 3.1.3.6, and (3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

SHUTDOWN MARGIN requirements vary throughout core life as a function of fuel depletion, RCS boron concentration, and RCS T_{cold}. The most restrictive condition occurs at EOL, with T_{cold} at no load operating temperature, and is associated with a postulated steam line break accident and resulting uncontrolled RCS cooldown. In the analysis of this accident, a minimum SHUTDOWN MARGIN of 6.0% delta k/k is required to control the reactivity transient. Accordingly, the SHUTDOWN MARGIN requirement is based upon this limiting condition and is consistent with the criteria used to establish the power dependent CEA insertion limits and with the assumptions used in the FSAR Safety Analysis.

With T_{cold} less than or equal to 210°F, the reactivity transients resulting from uncontrolled RCS cooldown are minimal and a 4% Δk/k SHUTDOWN MARGIN requirement is set to ensure that reactivity transients resulting from an inadvertent single CEA withdrawal event are minimal.

3/4.1.1.3 MODERATOR TEMPERATURE COEFFICIENT (MTC)

The limitations on moderator temperature coefficient (MTC) are provided to ensure that the assumptions used in the accident and transient analysis remain valid through each fuel cycle. The surveillance requirements for measurement of the MTC during each fuel cycle are adequate to confirm the MTC value since this coefficient changes slowly due principally to the reduction in RCS boron concentration associated with fuel burnup. The confirmation that the measured MTC value is within its limit provides assurances that the coefficient will be maintained within acceptable values throughout each fuel cycle.

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3/4.1.1.4 MINIMUM TEMPERATURE FOR CRITICALITY

This specification ensures that the reactor will not be made critical with the Reactor Coolant System cold leg temperature less than 552°F. This limitation is required to ensure (1) the moderator temperature coefficient is within its analyzed temperature range, (2) the protective instrumentation is within its normal operating range, and (3) to ensure consistency with the FSAR safety analysis.

3/4.1.2 BORATION SYSTEMS

The boron injection system ensures that negative reactivity control is available during each mode of facility operation. The components required to perform this function include (1) borated water sources, (2) charging pumps, (3) separate flow paths, and (4) an emergency power supply from OPERABLE diesel generators. The nominal capacity of each charging pump is 44 gpm at its discharge. Up to 16 gpm of this may be diverted to the volume control tank via the RCP control bleedoff. Instrument inaccuracies and pump performance uncertainties are limited to 2 gpm yielding the 26 gpm value.

With the RCS temperature above 210°F, a minimum of two separate and redundant boron injection systems are provided to ensure single functional capability in the event an assumed failure renders one of the systems inoperable. Allowable out-of-service periods ensure that minor component repair or corrective action may be completed without undue risk to overall facility safety from injection system failures during the repair period.

The boration capability of either system is sufficient to provide a SHUTDOWN MARGIN from expected operating conditions of 4% delta k/k after xenon decay and cooldown to 210°F. The maximum expected boration capability requirement occurs at EOL from full power equilibrium xenon conditions and requires ** ~~22,800~~ gallons of 4000 ppm borated water from either the refueling water tank or the spent fuel pool.

With the RCS temperature below 210°F one injection system is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the additional restrictions prohibiting CORE ALTERATIONS and positive reactivity changes in the event the single injection system becomes inoperable. The restrictions of one and only one operable charging pump whenever reactor coolant level is below the bottom of the pressurizer is based on the assumptions used in the analysis of the boron dilution event.

The boron capability required below 210°F is based upon providing a 4% delta k/k SHUTDOWN MARGIN after xenon decay and cooldown from 210°F to 120°F. This condition requires ~~9,700~~ ** gallons of 4000 ppm borated water from either the refueling water tank or the spent fuel pool.

** See Applicant's SAR

BORATION SYSTEMS (Continued)

The values of water volumes, temperatures, and boron concentration in the refueling water tank are provided to ensure that the assumptions used in the initial conditions of the LOCA Safety Analysis remain valid.

The OPERABILITY of one boron injection system during REFUELING ensures that this system is available for reactivity control while in MODE 6.

With the RCS temperature below 210°F while in MODES 5 and 6, a source of borated water is required to be available for reactivity control and makeup for losses due to contraction and evaporation. The requirement of 33,500 gallons of 4000 ppm borated water in either the refueling water tank or spent fuel pool ensures that this source is available.

~~The limits on contained water volume and boron concentration of the RWT also ensure a pH value of between 7.0 and 8.5 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components.~~

3/4.1.2.7 BORON DILUTION ALARMS

The startup channel high neutron flux alarms alert the operator to an inadvertent boron dilution. Both channels must be operating to assure detection of a boron dilution event by the high neutron flux alarms. If one or both of the alarms are inoperable at any time, the bases for ACTION statements are as follows:

a. One startup channel high neutron flux alarm not operating:

With only one startup channel high neutron flux alarm OPERABLE while in MODE ~~3,~~ 4, 5, or 6, a single failure to the alarm could prevent detection of boron dilution. By periodic monitoring of the RCS boron concentration by either boronometer or RCS sampling, a decrease in the boron concentration during an inadvertent boron dilution event will be observed. This provides alternate methods of detection of boron dilution with sufficient time for termination of the event before complete loss of SHUTDOWN MARGIN and return to criticality.

b. Both startup channel high neutron flux alarms not operating:

When both startup channel high neutron flux alarms are inoperable, there is no means of alarming on high neutron flux when subcritical. Therefore, either simultaneous use of the boronmeter and RCS sampling or independent collection and analysis of two RCS samples to monitor the RCS boron concentration provides alternate indications of inadvertent boron dilution. This will allow detection with sufficient time for termination of boron dilution before complete loss of shutdown margin and return to criticality.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

The specifications of this section ensure that (1) acceptable power distribution limits are maintained, (2) the minimum SHUTDOWN MARGIN is maintained, and (3) the potential effects of CEA misalignments are limited to acceptable levels.

The ACTION statements which permit limited variations from the basic requirements are accompanied by additional restrictions which ensure that the original design criteria are met.

The ACTION statements applicable to a stuck or untrippable CEA, to two or more inoperable CEAs, and to a large misalignment (greater than or equal to 19 inches) of two or more CEAs, require a prompt shutdown of the reactor since either of these conditions may be indicative of a possible loss of mechanical functional capability of the CEAs and in the event of a stuck or untrippable CEA, the loss of SHUTDOWN MARGIN.

For small misalignments (less than 19 inches) of the CEAs, there is (1) a small effect on the time-dependent long-term power distributions relative to those used in generating LCOs and LSSS setpoints, (2) a small effect on the available SHUTDOWN MARGIN, and (3) a small effect on the ejected CEA worth used in the safety analysis. Therefore, the ACTION statement associated with small misalignments of CEAs permits a 1-hour time interval during which attempts may be made to restore the CEA to within its alignment requirements. The 1-hour time limit is sufficient to (1) identify causes of a misaligned CEA, (2) take appropriate corrective action to realign the CEAs, and (3) minimize the effects of xenon redistribution.

The CPCs provide protection to the core in the event of a large misalignment (greater than or equal to 19 inches) of a CEA by applying appropriate penalty factors to the calculation to account for the misaligned CEA. However, this misalignment would cause distortion of the core power distribution. This distribution may, in turn, have a significant effect on (1) the available SHUTDOWN MARGIN, (2) the time-dependent long-term power distributions relative to those used in generating LCOs and LSSS setpoints, and (3) the ejected CEA worth used in the safety analysis. Therefore, the ACTION statement associated with the large misalignment of a CEA requires a prompt realignment of the misaligned CEA.

The ACTION statements applicable to misaligned or inoperable CEAs include requirements to align the OPERABLE CEAs in a given group with the inoperable CEA. Conformance with these alignment requirements bring the core, within a short period of time, to a configuration consistent with that assumed in generating LCO and LSSS setpoints. However, extended operation with CEAs significantly inserted in the core may lead to perturbations in (1) local burnup, (2) peaking factors, and (3) available SHUTDOWN MARGIN which are more adverse than the conditions assumed to exist in the safety analyses and LCO

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BASES

MOVABLE CONTROL ASSEMBLIES (Continued)

and LSSS setpoints determination. Therefore, time limits have been imposed on operation with inoperable CEAs to preclude such adverse conditions from developing.

Operability of at least two CEA position indicator channels is required to determine CEA positions and thereby ensure compliance with the CEA alignment and insertion limits. The CEA "Full In" and "Full Out" limits provide an additional independent means for determining the CEA positions when the CEAs are at either their fully inserted or fully withdrawn positions. Therefore, the ACTION statements applicable to inoperable CEA position indicators permit continued operations when the positions of CEAs with inoperable position indicators can be verified by the "Full In" or "Full Out" limits.

CEA positions and OPERABILITY of the CEA position indicators are required to be verified on a nominal basis of once per 12 hours with more frequent verifications required if an automatic monitoring channel is inoperable. These verification frequencies are adequate for assuring that the applicable LCOs are satisfied.

The maximum CEA drop time restriction is consistent with the assumed CEA drop time used in the safety analyses. Measurement with T_{cold} greater than or equal to 552°F and with all reactor coolant pumps operating ensures that the measured drop times will be representative of insertion times experienced during a reactor trip at operating conditions.

Several design steps were employed to accommodate the possible CEA guide tube wear which could arise from CEA vibrations when fully withdrawn. Specifically, a programmed insertion schedule will be used to cycle the CEAs between the full out position ("FULL OUT" LIMIT) and 3.0 inches inserted over the fuel cycle. This cycling will distribute the possible guide tube wear over a larger area, thus minimizing any effects. To accommodate this programmed insertion schedule, the fully withdrawn position was redefined, in some cases, to be 144.75 inches (193 steps) ~~or greater~~.

The establishment of LSSS and LCOs requires that the expected long- and short-term behavior of the radial peaking factors be determined. The long-term behavior relates to the variation of the steady-state radial peaking factors with core burnup and is affected by the amount of CEA insertion assumed, the portion of a burnup cycle over which such insertion is assumed and the expected power level variation throughout the cycle. The short-term behavior relates to transient perturbations to the steady-state radial peaks due to radial xenon redistribution. The magnitudes of such perturbations depend upon the expected use of the CEAs during anticipated power reductions

MOVABLE CONTROL ASSEMBLIES (Continued)

and load maneuvering. Analyses are performed based on the expected mode of operation of the NSSS (base loaded, or load maneuvering, etc.) and from these analyses CEA insertions are determined and a consistent set of radial peaking factors defined. The Long Term Steady State and Short Term Insertion Limits are determined based upon the assumed mode of operation used in the analyses and provide a means of preserving the assumptions on CEA insertions used. The limits specified serve to limit the behavior of the radial peaking factors within the bounds determined from analysis. The actions specified serve to limit the extent of radial xenon redistribution effects to those accommodated in the analyses. The Long and Short Term Insertion Limits of Specification 3.1.3.6 are specified for the plant which has been designed for primarily base loaded operation but which has the ability to accommodate a limited amount of load maneuvering.

The Transient Insertion Limits of Specification 3.1.3.6 and the Shutdown CEA Insertion Limits of Specification 3.1.3.5 ensure that (1) the minimum SHUTDOWN MARGIN is maintained, and (2) the potential effects of a CEA ejection accident are limited to acceptable levels. Long-term operation at the Transient Insertion Limits is not permitted since such operation could have effects on the core power distribution which could invalidate assumptions used to determine the behavior of the radial peaking factors.

The PVNGS CPC and COLSS systems are responsible for the safety and monitoring functions, respectively, of the reactor core. COLSS monitors the DNB Power Operating Limit (POL) and various operating parameters to help the operator maintain plant operation within the limiting conditions for operation (LCO). Operatting within the LCO guarantees that in the event of an Anticipated Operational Occurrence (AOO), the CPCs will provide a reactor trip in time to prevent unacceptable fuel damage.

The COLSS reserves the Required Overpower Margin (ROPM) to account for the Loss of Flow (LOF) transient which is the limiting AOO for the PVNGS plants. When the COLSS is Out of Service (COOS), the monitoring function is performed via the CPC calculation of DNBR in conjunction with a Technical Specification COOS Limit Line (Figure 3.2-2) which restricts the reactor power sufficiently to preserve the ROPM.

The reduction of the CEA deviation penalties in accordance with the CEAC (Control Element Assembly Calculator) sensitivity reduction program has been performed. This task involved setting many of the inward single CEA deviation penalty factors to 1.0. An inward CEA deviation event in effect would not be accompanied by the application of the CEA deviation penalty in either the CPC DNB and LHR (Linear Heat Rate) calculations for those CEAs with the reduced penalty factors. The protection for an inward CEA deviation event is thus accounted for separately.

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MOVABLE CONTROL ASSEMBLIES (Continued)

~~If an inward CEA deviation event occurs, the current CPC algorithm applies two penalty factors to each of the DNB and LHR calculations. The first, a static penalty factor, is applied upon detection of the event. The second, a xenon redistribution penalty, is applied linearly as a function of time after the CEA drop. The expected margin degradation for the inward CEA deviation event for which the penalty factor has been reduced is accounted for in two ways. The ROPM reserved in COLSS is used to account for some of the margin degradation. If the combination of the static and xenon redistribution penalties exceeds the reserved ROPM, a power reduction in accordance with the curve in Figure 3.1-2B is required. In addition, the part length CEA maneuvering is restricted in accordance with Figure 3.1-2A to justify reduction of the PLR deviation penalty factors.~~

~~The technical specification permits plant operation if both CEACs are considered inoperable for safety purposes after this period.~~

The Transient Insertion Limits of Specification 3.1.3.6 and the Shutdown CEA Insertion Limits of Specification 3.1.3.5 ensure that 1) the minimum SHUTDOWN MARGIN is maintained, and 2) the potential effects of a CEA ejection accident are limited to acceptable levels. Long term operation at the Transient Insertion Limits is not permitted since such operation could have effects on the core power distribution which could invalidate assumptions used to determine the behavior of the radial peaking factors.

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BASES

3/4.2.1 LINEAR HEAT RATE

The limitation on linear heat rate ensures that in the event of a LOCA, the peak temperature of the fuel cladding will not exceed 2200°F.

Either of the two core power distribution monitoring systems, the Core Operating Limit Supervisory System (COLSS) and the Local Power Density channels in the Core Protection Calculators (CPCs), provide adequate monitoring of the core power distribution and are capable of verifying that the linear heat rate does not exceed its limits. The COLSS performs this function by continuously monitoring the core power distribution and calculating a core power operating limit corresponding to the allowable peak linear heat rate. Reactor operation at or below this calculated power level assures that the limits of 14.0 kW/ft are not exceeded.

The COLSS calculated core power and the COLSS calculated core power operating limits based on linear heat rate are continuously monitored and displayed to the operator. A COLSS alarm is annunciated in the event that the core power exceeds the core power operating limit. This provides adequate margin to the linear heat rate operating limit for normal steady-state operation. Normal reactor power transients or equipment failures which do not require a reactor trip may result in this core power operating limit being exceeded. In the event this occurs, COLSS alarms will be annunciated. If the event which causes the COLSS limit to be exceeded results in conditions which approach the core safety limits, a reactor trip will be initiated by the Reactor Protective Instrumentation. The COLSS calculation of the linear heat rate includes appropriate penalty factors which provide, with a 95/95 probability/confidence level, that the maximum linear heat rate calculated by COLSS is conservative with respect to the actual maximum linear heat rate existing in the core. These penalty factors are determined from the uncertainties associated with planar radial peaking measurement, engineering heat flux uncertainty, axial densification, software algorithm modelling, computer processing, rod bow, and core power measurement.

Parameters required to maintain the operating limit power level based on linear heat rate, margin to DNB, and total core power are also monitored by the CPCs (assuming minimum core power of 20% of RATED THERMAL POWER). The 20% RATED THERMAL POWER threshold is due to the neutron flux detector system being inaccurate below 20% core power. Core noise level at low power is too large to obtain usable detector readings. Therefore, in the event that the COLSS is not being used, operation within the limits of Figure 3.2-2 can be maintained by utilizing a predetermined local power density margin and a total core power limit in the CPC trip channels. The above listed uncertainty and penalty factors plus those associated with the CPC startup test acceptance criteria are also included in the CPCs.

3/4.2.2 PLANAR RADIAL PEAKING FACTORS

Limiting the values of the PLANAR RADIAL PEAKING FACTORS (F_{xy}^{c}) used in the COLSS and CPCs to values equal to or greater than the measured PLANAR RADIAL PEAKING FACTORS (F_{xy}^{m}) provides assurance that the limits calculated by COLSS and the CPCs remain valid. Data from the incore detectors are used for determining the measured PLANAR RADIAL PEAKING FACTORS. A minimum core power at 20% of RATED THERMAL POWER is assumed in determining the PLANAR RADIAL PEAKING FACTORS. The 20% RATED THERMAL POWER threshold is due to the neutron flux detector system being inaccurate below 20% core power. Core noise level at low power is too large to obtain usable detector readings. The periodic surveillance requirements for determining the measured PLANAR RADIAL PEAKING FACTORS provides assurance that the PLANAR RADIAL PEAKING FACTORS used in COLSS and the CPCs remain valid throughout the fuel cycle. Determining the measured PLANAR RADIAL PEAKING FACTORS after each fuel loading prior to exceeding 70% of RATED THERMAL POWER provides additional assurance that the core was properly loaded.

3/4.2.3 AZIMUTHAL POWER TILT - T_q

The limitations on the AZIMUTHAL POWER TILT are provided to ensure that design safety margins are maintained. An AZIMUTHAL POWER TILT greater than 0.10 is not expected and if it should occur, operation is restricted to only those conditions required to identify the cause of the tilt. The tilt is normally calculated by COLSS. A minimum core power of 20% of RATED THERMAL POWER is assumed by the CPCs in its input to COLSS for calculation of AZIMUTHAL POWER TILT. The 20% RATED THERMAL POWER threshold is due to the neutron flux detector system being inaccurate below 20% core power. Core noise level at low power is too large to obtain usable detector readings. The surveillance requirements specified when COLSS is out of service provide an acceptable means of detecting the presence of a steady-state tilt. It is necessary to explicitly account for power asymmetries because the radial peaking factors used in the core power distribution calculations are based on an untilted power distribution.

The AZIMUTHAL POWER TILT is equal to $(P_{tilt}/P_{untilt})-1.0$ where:

AZIMUTHAL POWER TILT is measured by assuming that the ratio of the power at any core location in the presence of a tilt to the untilted power at the location is of the form:

$$P_{tilt}/P_{untilt} = 1 + T_q\, g \cos(\theta - \theta_0)$$

where:

T_q is the peak fractional tilt amplitude at the core periphery

g is the radial normalizing factor

θ is the azimuthal core location

θ_0 is the azimuthal core location of maximum tilt

AZIMUTHAL POWER TILT - T_q (Continued)

P_{tilt}/P_{untilt} is the ratio of the power at a core location in the presence of a tilt to the power at that location with no tilt.

~~The AZIMUTHAL POWER TILT allowance used in the CPCs is defined as the value of CPC addressable constant TR-1.0~~

3/4.2.4 DNBR MARGIN

The limitation on DNBR as a function of AXIAL SHAPE INDEX represents a conservative envelope of operating conditions consistent with the safety analysis assumptions and which have been analytically demonstrated adequate to maintain an acceptable minimum DNBR throughout all anticipated operational occurrences, of which the loss of flow transient is the most limiting. Operation of the core with a DNBR at or above this limit provides assurance that an acceptable minimum DNBR will be maintained in the event of a loss of flow transient.

Either of the two core power distribution monitoring systems, the Core Operating Limit Supervisory System (COLSS) and the DNBR channels in the Core Protection Calculators (CPCs), provide adequate monitoring of the core power distribution and are capable of verifying that the DNBR does not violate its limits. The COLSS performs this function by continuously monitoring the core power distribution and calculating a core operating limit corresponding to the allowable minimum DNBR. Reactor operation at or below this calculated power level assures that the limits of Figure 3.2-1 are not violated. The COLSS calculation of core power operating limit based on DNBR includes appropriate penalty factors which provide, with a 95/95 probability/confidence level, that the core power limits calculated by COLSS (based on the minimum DNBR Limit) is conservative with respect to the actual core power limit. These penalty factors are determined from the uncertainties associated with planar radial peaking measurement, engineering heat flux, state parameter measurement, software algorithm modelling, computer processing, rod bow, and core power measurement.

Parameters required to maintain the margin to DNB and total core power are also monitored by the CPCs. Therefore, in the event that the COLSS is not being used, operation within the limits of Figure 3.2-2 can be maintained by utilizing a predetermined DNBR as a function of AXIAL SHAPE INDEX and by monitoring the CPC trip channels. The above listed uncertainty and penalty factors are also included in the CPCs which assume a minimum core power of 20% of RATED THERMAL POWER. The 20% RATED THERMAL POWER threshold is due to the neutron flux detector system being inaccurate below 20% core power. Core noise level at low power is too large to obtain usable detector readings.

The DNBR penalty factors listed in Specification 4.2.4.4 are penalties used to accommodate the effects of rod bow. The amount of rod bow in each assembly is dependent upon the average burnup experienced by that assembly. Fuel assemblies that incur higher average burnup will experience a greater magnitude of rod bow. Conversely, lower burnup assemblies will experience less rod bow. The penalty for each batch required to compensate for rod bow is determined from a batch's maximum average assembly burnup applied to the batch's maximum integrated planar-radial power peak. A single net penalty for COLSS and CPC is then determined from the penalties associated with each batch, accounting for the offsetting margins due to the lower radial power peaks in the higher burnup batches.

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3/4.2.5 RCS FLOW RATE

This specification is provided to ensure that the actual RCS total flow rate is maintained at or above the minimum value used in the safety analyses.

3/4.2.6 REACTOR COOLANT COLD LEG TEMPERATURE

This specification is provided to ensure that the actual value of reactor coolant cold leg temperature is maintained within the range of values used in the safety analyses.

3/4.2.7 AXIAL SHAPE INDEX

This specification is provided to ensure that the actual value of the core average AXIAL SHAPE INDEX is maintained within the range of values used in the safety analyses.

3/4.2.8 PRESSURIZER PRESSURE

This specification is provided to ensure that the actual value of pressurizer pressure is maintained within the range of values used in the safety analyses.

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3/4.3.1 and 3/4.3.2 REACTOR PROTECTIVE AND ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

The OPERABILITY of the reactor protective and Engineered Safety Features Actuation Systems instrumentation and bypasses ensures that (1) the associated Engineered Safety Features Actuation action and/or reactor trip will be initiated when the parameter monitored by each channel or combination thereof reaches its setpoint, (2) the specified coincidence logic is maintained, (3) sufficient redundancy is maintained to permit a channel to be out of service for testing or maintenance, and (4) sufficient system functional capability is available from diverse parameters.

The OPERABILITY of these systems is required to provide the overall reliability, redundancy, and diversity assumed available in the facility design for the protection and mitigation of accident and transient conditions. The integrated operation of each of these systems is consistent with the assumptions used in the safety analyses.

Response time testing of resistance temperature devices, which are a part of the reactor protective system, shall be performed by using in-situ loop current test techniques or another NRC approved method.

Any modifications which are made to the core protection calculator software (including changes of algorithms and fuel cycle specific data) shall be performed in accordance with ~~"CPC Protection Algorithm Software Change Procedure," CEN-39(A)-P, Revisions 2 and Supplement 1-P, Revision 01 or another~~ an NRC approved procedure on CPC software modifications.

The design of the Control Element Assembly Calculators (CEAC) provides reactor protection in the event one or both CEACs become inoperable. If one CEAC is in test or inoperable, verification of CEA position is performed at least every 4 hours. If the second CEAC fails, the CPCs in conjunction with plant Technical Specifications will use DNBR and LPD penalty factors and increased DNBR and LPD margin to restrict reactor operation to a power level that will ensure safe operation of the plant. If the margins are not maintained, a reactor trip will occur.

~~The value of the DNBR in Specification 2.1 is conservatively compensated for measurement uncertainties. Therefore, the actual RCS total flow rate determined by the reactor coolant pump differential pressure instrumentation or by calorimetric calculations does not have to be conservatively compensated for measurement uncertainties.~~

~~An analysis was done to specify a minimum power level below which an additional power reduction is unnecessary even if there is a CEA misalignment with CEACs out of service.~~

~~This work is the completion of the CEAC's Out of Service (OOS) work. This analysis improves ANPP Unit 1, Cycle 1 power capability from about 75% to greater than about 90% with both CEACs out of service.~~

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REACTOR PROTECTIVE AND ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION (Continued)

~~The analysis determined a Power Operating Limit (POL) power and assumed a CEA misalignment occurred from this power level. The power penalty factor that would accommodate changes in radial peaks and one hour xenon redistribution that would occur if there were a CEA misalignment with CEACs out of service. The quotient of the POL power and the CEA misalignment Power Penalty factor is the maximum power (50% power) at which DNBR SAFDL violation will occur even if there is a CEA misalignment from POL conditions. Below this power, extra thermal margin will be available to the plant. Thus, for CEA misalignment, power reduction below this limiting power is unnecessary.~~

~~The lowest core power for a POL was calculated to be 70% of rated power. This was based on the following worst COLSS fluid conditions.~~

~~High Temperature~~	:	~~580°F~~
~~Low Pressure~~	:	~~1785 psia~~
~~ASI~~	:	~~-.3~~
~~Underflow fraction~~	:	~~0.865~~
~~Low Flow~~	:	~~95% of full flow~~
~~High Radial Peak~~	:	~~1.70 (Bank 5+4+PLR; PDIL = 40% Power)~~

The surveillance requirements specified for these systems ensure that the overall system functional capability is maintained comparable to the original design standards. The periodic surveillance tests performed at the minimum frequencies are sufficient to demonstrate this capability.

The measurement of response time at the specified frequencies provides assurance that the protective and ESF action function associated with each channel is completed within the time limit assumed in the safety analyses. No credit was taken in the analyses for those channels with response times indicated as not applicable. The response times in Table 3.3-2 are made up of the time to generate the trip signal at the detector (sensor response time) and the time for the signal to interrupt power to the CEA drive mechanism (signal or trip delay time). The response times are taken from the sequence-of-events Tables in Section 15 of CESSAR.

Response time may be demonstrated by any series of sequential, overlapping, or total channel test measurements provided that such tests demonstrate the total channel response time as defined. Sensor response time verification may be demonstrated by either (1) in place, onsite, or offsite test measurements or (2) utilizing replacement sensors with certified response times.

3/4.3.3 MONITORING INSTRUMENTATION

3/4.3.3.1 RADIATION MONITORING INSTRUMENTATION

TP See Applicant's SA

~~The OPERABILITY of the radiation monitoring channels ensures that (1) the radiation levels are continually measured in the areas served by the individual channels and (2) the alarm or automatic action is initiated when the radiation level trip setpoint is exceeded.~~

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3/4.3.3.2 INCORE DETECTORS

The OPERABILITY of the incore detectors with the specified minimum complement of equipment ensures that the measurements obtained from use of this system accurately represent the spatial neutron flux distribution of the reactor core.

3/4.3.3.3 SEISMIC INSTRUMENTATION

R See Applicant's SAR.

~~The OPERABILITY of the seismic instrumentation ensures that sufficient capability is available to promptly determine the magnitude of a seismic event and evaluate the response of those features important to safety. This capability is required to permit comparison of the measured response to that used in the design basis for the facility to determine if plant shutdown is required pursuant to Appendix A of 10 CFR Part 100. The instrumentation is consistent with the recommendations of Regulatory Guide 1.12, "Instrumentation for Earthquakes," April 1974 as identified in the PVNGS FSAR.~~

3/4.3.3.4 METEOROLOGICAL INSTRUMENTATION

R See Applicant's SAR

~~The OPERABILITY of the meteorological instrumentation ensures that sufficient meteorological data are available for estimating potential radiation doses to the public as a result of routine or accidental release of radioactive materials to the atmosphere. This capability is required to evaluate the need for initiating protective measures to protect the health and safety of the public and is consistent with the recommendations of Regulatory Guide 1.23 "Onsite Meteorological Programs," February 1972. Wind speeds less than 0.6 MPH cannot be measured by the meteorological instrumentation.~~

3/4.3.3.5 REMOTE SHUTDOWN SYSTEM INSTRUMENTATION

The OPERABILITY of the remote shutdown system instrumentation ensures that sufficient capability is available to permit safe shutdown and maintenance of HOT STANDBY of the facility from locations outside of the control room. This capability is required in the event control room habitability is lost and is consistent with General Design Criterion 19 of 10 CFR Part 50.

The parameters selected to be monitored ensure that (1) the condition of the reactor is known, (2) conditions in the RCS are known, (3) the steam generators are available for residual heat removal, (4) a source of water is available for makeup to the RCS, and (5) the charging system is available to makeup water to the RCS.

The OPERABILITY of the remote shutdown system insures that a fire will not preclude achieving safe shutdown. The remote shutdown system instrumentation, and control [redacted] circuits [redacted] necessary to eliminate effects of the fire and allow operation of instrumentation, control and power circuits required to achieve and maintain a safe shutdown condition are independent of areas where a fire could damage systems normally used to shutdown the reactor. This capability is consistent with General Design Criterion 3 and Appendix R to 10 CFR 50.

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REMOTE SHUTDOWN SYSTEM (Continued)

~~The alternate disconnect methods of power or control circuits ensure that sufficient capability is available to permit shutdown and maintenance of cold shutdown of the facility by relying on additional operator actions at local control stations rather than at the RSP.~~

3/4.3.3.6 POST-ACCIDENT MONITORING INSTRUMENTATION

The OPERABILITY of the post-accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables following an accident. This capability is consistent with the recommendations of Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Plants to Assess Plant Conditions During and Following an Accident," December 1975 and NUREG 0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations."

~~The containment high range area monitors (RU-148 & RU-149) and the main steamline radiation monitors (RU-139 A&B and RU-140 A&B) are in Table 3.3-6. The high range effluent monitors and samplers (RU-142, RU-144 and RU-146) are in Table 3.3-13. The containment hydrogen monitors are in Specification 3/4.6.5.1. The Post Accident Sampling System (RCS coolant) is in Table 3.3-6.~~

The Subcooled Margin Monitor (SMM), the Heat Junction Thermocouple (HJTC), and the Core Exit Thermocouples (CET) comprise the Inadequate Core Cooling (ICC) instrumentation required by Item II.F.2 NUREG-0737, the Post TMI-2 Action Plan. The function of the ICC instrumentation is to enhance the ability of the plant operator to diagnose the approach to existance of, and recovery from ICC. Additionally, they aid in tracking reactor coolant inventory. These instruments are included in the Technical Specifications at the request of NRC Generic Letter 83-37. These are not required by the accident analysis, nor to bring the plant to Cold Shutdown.

In the event more than four sensors in a Reactor Vessel Level channel are inoperable, repairs may only be possible during the next refueling outage. This is because the sensors are accessible only after the missile shield and reactor vessel head are removed. It is not feasible to repair a channel except during a refueling outage when the missile shield and reactor vessel head are removed to refuel the core. If both channels are inoperable, the channels shall be restored to OPERABLE status in the nearest refueling outage. If only one channel is inoperable, it is intented that this channel be restored to OPERABLE status in a refueling outage as soon as reasonably possible.

Insert

3/4.3.3.8 FIRE DETECTION INSTRUMENTATION

See Applicant's SAR

~~OPERABILITY of the fire detection instrumentation ensures that adequate warning capability is available for the prompt detection of fires and that fire suppression systems, that are actuated by fire detectors, will discharge~~

Insert

3/4.3.3.7 CHLORINE DETECTION SYSTEMS

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FIRE DETECTION INSTRUMENTATION (Continued)

~~extinguishing agent in a timely manner. Prompt detection and suppression of fires will reduce the potential for damage to safety-related equipment and is an integral element in the overall facility fire protection program.~~

~~Fire detectors that are used to actuate fire suppression systems represent a more critically important component of a plant's fire protection program than detectors that are installed solely for early fire warning and notification. Consequently, the minimum number of operable fire detectors must be greater.~~

~~The loss of detection capability for fire suppression systems, actuated by fire detectors, represents a significant degradation of fire protection for any area. As a result, the establishment of a fire watch patrol must be initiated at an earlier stage than would be warranted for the loss of detectors that provide only early fire warning. The establishment of frequent fire patrols in the affected areas is required to provide detection capability until the inoperable instrumentation is restored to OPERABILITY.~~

~~The fire zones listed in Table 3.3-11, Fire Detection Instruments, are discussed in Section 9B of the PVNGS FSAR.~~

3/4.3.3.~~8~~ 9 LOOSE-PART DETECTION INSTRUMENTATION

~~The OPERABILITY of the loose-part detection instrumentation ensures that sufficient capability is available to detect loose metallic parts in the primary system and avoid or mitigate damage to primary system components. The allowable out-of-service times and surveillance requirements are consistent with the recommendations of Regulatory Guide 1.133, "Loose-Part Detection Program for the Primary System of Light-Water-Cooled Reactors," May 1981.~~

3/4.3.3.~~9~~ 10 RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

~~The radioactive gaseous effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in gaseous effluents during actual or potential releases of gaseous effluents. The alarm/trip setpoints for these instruments shall be calculated and adjusted in accordance with the methodology and parameters in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. This instrumentation also includes provisions for monitoring (and controlling) the concentrations of potentially explosive gas mixtures in the GASEOUS RADWASTE SYSTEM. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63, and 64 of Appendix A to 10 CFR Part 50.~~

~~There are two separate radioactive gaseous effluent monitoring systems: the low range effluent monitors for normal plant radioactive gaseous effluents and the high range effluent monitors for post-accident plant radioactive gaseous effluents. The low range monitors operate at all times until the concentration of radioactivity in the effluent becomes too high during post-accident conditions. The high range monitors only operate when the concentration of radioactivity in the effluent is above the setpoint in the low range monitors.~~

Insert

(see Applicant's SAR)

Insert

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3/4.3.4 TURBINE OVERSPEED PROTECTION

See Applicant's SAR.

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3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

The plant is designed to operate with both reactor coolant loops and associated reactor coolant pumps in operation, and maintain DNBR above 1.231 during all normal operations and anticipated transients. In MODES 1 and 2 with one reactor coolant loop not in operation, this specification requires that the plant be in at least HOT STANDBY within 1 hour.

In MODE 3, a single reactor coolant loop provides sufficient heat removal capability for removing decay heat; however, single failure considerations require that two loops be OPERABLE.

In MODE 4, and in MODE 5 with reactor coolant loops filled, a single reactor coolant loop or shutdown cooling loop provides sufficient heat removal capability for removing decay heat; but single failure considerations require that at least two loops (either shutdown cooling or RCS) be OPERABLE. Thus, if the reactor coolant loops are not OPERABLE, this specification requires that two shutdown cooling loops be OPERABLE.

In MODE 5 with reactor coolant loops not filled, a single shutdown cooling loop provides sufficient heat removal capability for removing decay heat; but single failure considerations, and the unavailability of the steam generators as a heat removing component, require that at least two shutdown cooling loops be OPERABLE.

The operation of one reactor coolant pump or one shutdown cooling pump provides adequate flow to ensure mixing, prevent stratification, and produce gradual reactivity changes during boron concentration reductions in the Reactor Coolant System. A flow rate of at least 4000 gpm will circulate one equivalent Reactor Coolant System volume of 12,097 cubic feet in approximately 23 minutes. The reactivity change rate associated with boron reductions will, therefore, be within the capability of operator recognition and control.

The restrictions on starting a reactor coolant pump in MODES 4 and 5, ** with one or more RCS cold legs less than or equal to ~~[illegible]~~°F during cooldown or ~~[illegible]~~°F during heatup are provided to prevent RCS pressure transients, caused by energy additions from the secondary system, which could exceed the limits of Appendix G to 10 CFR Part 50. The RCS will be protected against overpressure transients and will not exceed the limits of Appendix G by restricting starting of the RCPs to when the secondary water temperature of each steam generator is less than 100°F above each of the RCS cold leg temperatures.

3/4.4.2 SAFETY VALVES

The pressurizer code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2750 psia. Each safety valve is designed to relieve a minimum of 460,000 lb per hour of saturated steam ~~at the valve setpoint~~. The relief capacity of a single safety valve is adequate to relieve any overpressure condition which could occur during shutdown. In the event that no safety valves are OPERABLE, an operating shutdown cooling loop, connected to the RCS, provides overpressure relief capability and will prevent RCS overpressurization.

** See Applicant's SAR.

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SAFETY VALVES (Continued)

During operation, all pressurizer code safety valves must be OPERABLE to prevent the RCS from being pressurized above its Safety Limit of 2750 psia. The combined relief capacity of these valves is sufficient to limit the system pressure to within its Safety Limit of 2750 psia following a complete loss of turbine generator load while operating at RATED THERMAL POWER and assuming no reactor trip until the first Reactor Protective System trip setpoint (Pressurizer Pressure-High) is reached (i.e., ~~there is no~~ no credit is taken for a direct reactor trip on the loss of turbine) and also assuming no operation of the steam dump valves.

Demonstration of the safety valves' lift settings will occur only during shutdown and will be performed in accordance with the provisions of Section XI of the ASME Boiler and Pressure Vessel Code.

3/4.4.3 PRESSURIZER

An OPERABLE pressurizer provides pressure control for the Reactor Coolant System during operations with both forced reactor coolant flow and with natural circulation flow. The minimum water level in the pressurizer assures the pressurizer heaters, which are required to achieve and maintain pressure control, remain covered with water to prevent failure, which could occur if the heaters were energized uncovered. The maximum water level in the pressurizer ensures that this parameter is maintained within the envelope of operation assumed in the safety analysis. The maximum water level also ensures that the RCS is not a hydraulically solid system and that a steam bubble will be provided to accommodate pressure surges during operation. The steam bubble also protects the pressurizer code safety valves against water relief. The requirement to verify that on an Engineered Safety Features Actuation test signal concurrent with a loss-of-offsite power the pressurizer heaters are automatically shed from the emergency power sources is to ensure that the non-Class 1E heaters do not reduce the reliability of or overload the emergency power source. The requirement that a minimum number of pressurizer heaters be OPERABLE enhances the capability to control Reactor Coolant System pressure and establish and maintain natural circulation.

The auxiliary pressurizer spray is required to depressurize the RCS by cooling the pressurizer steam space to permit the plant to enter shutdown cooling. The auxiliary pressurizer spray is required during those periods when normal pressurizer spray is not available, such as during natural circulation and during the later stages of a normal RCS cooldown. The auxiliary pressurizer spray also distributes boron to the pressurizer when normal pressurizer spray is not available. Use of the auxiliary pressurizer spray is required during the recovery from a steam generator tube rupture and a small loss of coolant accident.

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3/4.4.4 STEAM GENERATORS

The surveillance requirements for inspection of the steam generator tubes ensure that the structural integrity of this portion of the RCS will be maintained. The program for inservice inspection of steam generator tubes is based on a modification of Regulatory Guide 1.83, Revision 1. Inservice inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes in the event that there is evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or inservice conditions that lead to corrosion.

Inservice inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those chemistry limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these limits, localized corrosion may likely result in stress corrosion cracking. The extent of cracking during plant operation would be limited by the limitation of steam generator tube leakage between the primary coolant system and the secondary coolant system (primary-to-secondary leakage = 0.5 gpm per steam generator). Cracks having a primary-to-secondary leakage less than this limit during operation will have an adequate margin of safety to withstand the loads imposed during normal operation and by postulated accidents. Operating plants have demonstrated that primary-to-secondary leakage of 0.5 gpm per steam generator can readily be detected by radiation monitors of steam generator blowdown. Leakage in excess of this limit will require plant shutdown and an unscheduled inspection, during which the leaking tubes will be located and plugged.

Wastage-type defects are unlikely with proper chemistry treatment of the ~~secondary~~ coolant. However, even if a defect should develop in service, it will be found during scheduled inservice steam generator tube examinations. Plugging will be required for all tubes with imperfections exceeding the plugging limit of 40% of the tube nominal wall thickness. Steam generator tube inspections of operating plants have demonstrated the capability to reliably detect degradation that has penetrated 20% of the original tube wall thickness.

Whenever the results of any steam generator tubing inservice inspection fall into Category C-3, these results will be promptly reported to the Commission pursuant to Specification 6.9.1 prior to the resumption of plant operation. Such cases will be considered by the Commission on a case-by-case basis and may result in a requirement for analysis, laboratory examinations, tests, additional eddy-current inspection, and revision of the Technical Specifications, if necessary.

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3/4.4.5 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.5.1 LEAKAGE DETECTION SYSTEMS

R See Applicant's SAR.

~~The RCS leakage detection systems required by this specification are provided to monitor and detect leakage from the reactor coolant pressure boundary. Containment sump flow is provided by monitoring the rate of sump level increase prior to the sump being pumped down, and is alarmed at the equivalent of 1 gpm leakage into the sump. These detection systems are consistent with the recommendations of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," May 1973.~~

3/4.4.5.2 OPERATIONAL LEAKAGE

Industry experience has shown that while a limited amount of leakage is expected from the RCS, the unidentified portion of this leakage can be reduced to a threshold value. A threshold value of less than 1 gpm is sufficiently low to ensure early detection of additional leakage.

The 10 gpm IDENTIFIED LEAKAGE limitation provides allowances for a limited amount of leakage from known sources whose presence will not interfere with the detection of UNIDENTIFIED LEAKAGE by the leakage detection systems.

The surveillance requirements for RCS pressure isolation valves provide added assurance of valve integrity thereby reducing the probability of valve failure and consequent intersystem LOCA. Leakage from the RCS pressure isolation valves is IDENTIFIED LEAKAGE and will be considered as a portion of the allowable limit.

The total steam generator tube leakage limit of 1 gpm for both steam generators ensures that the dosage contribution from the tube leakage will be limited to less than Part 100 guidelines for infrequent and limiting fault events. The 1 gpm limit is consistent with the assumptions used in the analysis of these accidents. ~~Section 15.4.1 of the PVNGS SER dated November 11, 1981, stated that the primary-to-secondary leakage from the steam generators should be less than or equal to 0.5 gpm. This was based on the bounding accident analysis in Section 15 of CESSAR. The PVNGS meteorological parameters are sufficiently less than the parameters assumed in CESSAR to allow the Limiting Condition for Operation to be 1 gpm (instead of the 0.3 gpm) total primary-to-secondary leakage through all steam generators and 720 gallons per day through any one steam generator.~~ The ~~[illegible]~~ 0.5 gpm leakage limit per steam generator ensures that steam generator tube integrity is maintained in the event of a main steam line rupture or under LOCA conditions.

PRESSURE BOUNDARY LEAKAGE of any magnitude may be indicative of an impending failure of the pressure boundary. Therefore, the presence of any PRESSURE BOUNDARY LEAKAGE requires the unit to be promptly placed in COLD SHUTDOWN.

** See Applicant's SAR.

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3/4.4.6 CHEMISTRY

The limitations on Reactor Coolant System chemistry ensure that corrosion of the Reactor Coolant System is minimized and reduces the potential for Reactor Coolant System leakage or failure due to stress corrosion. Maintaining the chemistry within the Steady State Limits provides adequate corrosion protection to ensure the structural integrity of the Reactor Coolant System over the life of the plant. The associated effects of exceeding the oxygen, chloride, and fluoride limits are time and temperature dependent. Corrosion studies show that operation may be continued with contaminant concentration levels in excess of the Steady State Limits, up to the Transient Limits, for the specified limited time intervals without having a significant effect on the structural integrity of the Reactor Coolant System. The time interval permitting continued operation within the restrictions of the Transient Limits provides time for taking corrective actions to restore the contaminant concentrations to within the Steady State Limits.

The surveillance requirements provide adequate assurance that concentrations in excess of the limits will be detected in sufficient time to take corrective action.

3/4.4.7 SPECIFIC ACTIVITY

The limitations on the specific activity of the primary coolant ensure that the resulting 2-hour doses at the site boundary will not exceed an appropriately small fraction of Part 100 limits 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 and a concurrent loss-of-offsite electrical power. The values for the limits on specific activity represent limits based upon a parametric evaluation by the NRC of typical site locations. ** These values are conservative in that specific site parameters of the ~~Palo Verde~~ site, such as site boundary location and meteorological conditions, were not considered in this evaluation.

The ACTION statement permitting POWER OPERATION to continue for limited time periods with the primary coolant's specific activity greater than 1.0 microcurie/gram DOSE EQUIVALENT I-131, but within the allowable limit shown on Figure 3.4-1, accommodates possible iodine spiking phenomenon which may occur following changes in THERMAL POWER. Operation with specific activity levels exceeding 1.0 microcurie/gram DOSE EQUIVALENT I-131 but within the limits shown on Figure 3.4-1 must be restricted to no more than 800 hours per year (approximately 10% of the unit's yearly operating time) since the activity levels allowed by Figure 3.4-1 increase the 2-hour thyroid dose at the site boundary by a factor of up to 20 following a postulated steam generator tube rupture. The reporting of cumulative operating time over 500 hours in any 6 month consecutive period with greater than 1.0 microcurie/gram DOSE EQUIVALENT I-131 will allow sufficient time for Commission evaluation of the circumstances prior to reaching the 800-hour limit.

** See Applicant's SAR

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SPECIFIC ACTIVITY (Continued)

Reducing T_{cold} to less than 500°F prevents the release of activity should a steam generator tube rupture since the saturation pressure of the primary 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 primary coolant will be detected in sufficient time to take corrective action. Information obtained on iodine spiking will be used to assess the parameters associated with spiking phenomena. A reduction in frequency of isotopic analyses following power changes may be permissible if justified by the data obtained.

3/4.4.8 PRESSURE/TEMPERATURE LIMITS

All components in the Reactor Coolant System are designed to withstand the effects of cyclic loads due to system temperature and pressure changes. These cyclic loads are introduced by normal load transients, reactor trips, and startup and shutdown operations. The various categories of load cycles used for design purposes are provided in Chapters 3 and 5 of the FSAR. During startup and shutdown, the rates of temperature and pressure changes are limited so as not to exceed the limit lines of Figure 3.4-2. This ensures that the maximum specified heatup and cooldown rates are consistent with the design assumptions and satisfy the stress limits for cyclic operation.

During heatup, the thermal gradients in the reactor vessel wall produce thermal stresses which vary from compressive at the inner wall to tensile at the outer wall. These thermal induced compressive stresses at the inner wall tend to alleviate the tensile stresses induced by the internal pressure.

At the outer wall of the vessel, these thermal stresses are additive to the pressure induced tensile stresses. The magnitude of the thermal stresses at either location is dependent on the rate of heatup. Consequently, each heatup rate of interest must be analyzed on an individual basis for both the inner and outer wall.

The heatup and cooldown limit curves (Figures 3.4-2 ~~and 3.4-3~~) are composite curves which were prepared by determining the most conservative case, with either the inside or outside wall controlling, for any heatup or cooldown rates of up to 100°F per hour. The heatup and cooldown curves were prepared based upon the most limiting value of the predicted adjusted reference temperature at the end of the service period indicated on Figures 3.4-2 ~~and 3.4-3~~. X

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PRESSURE/TEMPERATURE LIMITS (Continued)

The reactor vessel materials have been tested to determine their initial RT_{NDT}; the results of these test are shown in Table B 3/4.4-1. Reactor operation and resultant fast neutron (E greater than 1 MeV) irradiation will cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature, based upon the fluence and residual element content, can be predicted using Figure B 3/4.4-1 and the recommendations of Regulatory Guide 1.99, Revision 1, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials." The heatup and cooldown limit curves Figure~~s~~ 3.4-2 ~~and 3.4-3~~ include predicted adjustments for this shift in RT_{NDT} at the end of the applicable service period, as well as adjustments for possible errors in the pressure and temperature sensing instruments. X

The actual shift in RT_{NDT} of the vessel material will be established periodically during operation by removing and evaluating, in accordance with ASTM E185-73 and Appendix H of 10 CFR 50, reactor vessel material irradiation surveillance specimens installed near the inside wall of the reactor vessel in the core area. Since the neutron spectra at the irradiation samples and vessel inside radius are essentially identical, the measured transition shift for a sample can be applied with confidence to the adjacent section of the reactor vessel. The heatup and cooldown curves must be recalculated when the delta RT_{NDT} determined from the surveillance capsule is different from the calculated delta RT_{NDT} for the equivalent capsule radiation exposure.

The pressure-temperature limit lines shown on Figure 3.4-2 for reactor criticality and for inservice leak and hydrostatic testing have been provided to assure compliance with the minimum temperature requirements of Appendix G to 10 CFR Part 50. The reactor vessel material irradiation surveillance specimens are removed and examined to determine changes in material properties. The results of these examinations shall be used to update Figure 3.4-2 based on the greater of the following:

(1) the actual shift in reference for ~~[illegible]~~ (Table B 3/4 4 as determined by impact testing, or

(2) the predicted shift in reference temperature for the limiting weld and plate as determined by RG 1.99, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials."

The maximum RT_{NDT} for all Reactor Coolant System pressure-retaining materials, with the exception of the reactor pressure vessel, has been determined to be 40°F. The Lowest Service Temperature limit is based upon this RT_{NDT} since Article NB-2332 (Summer Addenda of 1972) of Section III of the ASME Boiler and Pressure Vessel Code requires the Lowest Service Temperature to be RT_{NDT} + 100°F for piping, pumps, and valves. Below this temperature, the system pressure must be limited to a maximum of 20% of the system's hydrostatic test pressure of 3125 psia. ~~However, based upon the 10 CFR Part 50 Appendix G analysis, the isothermal condition for the reactor vessel is more restrictive than the Lowest Service Temperature line. Therefore, only the isothermal line is shown on Figure 3.4-2.~~

The number of reactor vessel irradiation surveillance specimens and the frequencies for removing and testing these specimens are provided in Table 4.4-3 to assure compliance with the requirements of Appendix H to 10 CFR Part 50.

See Applicants' SAR

~~PALO VERDE - UNIT 1~~
CESSAR80-A1SSS-STS

TABLE B 3/4.4-1

REACTOR VESSEL TOUGHNESS

(FORGINGS)

PIECE NO.	CODE NO.	MATERIAL	VESSEL LOCATION	DROP WEIGHT RESULTS (°F)	RT_{NDT} (a)(b) (°F)	TEMPERATURE OF CHARPY V-NOTCH* @ 30 ft - lb	TEMPERATURE OF CHARPY V-NOTCH* @ 50 ft - lb	MINIMUM UPPER SHELF C_v ENERGY FOR LONGITUDINAL DIRECTION-ft lb
120-101	M-6703-1	SA 508-CL2	Inlet Nozzle	20	0	+20	+60	N.A.
120-101	M-6703-2	SA 508-CL2	Inlet Nozzle	+10	+10	-25	+10	N.A.
120-101	M-6703-3	SA 508-CL2	Inlet Nozzle	10	-10	-27	+18	N.A.
120-101	M-6703-4	SA 508-CL2	Inlet Nozzle	0	0	+5	+42	N.A.
131-102	M-4307-1	SA 508-CL2	Outlet Nozzle Safe End	10	+10	+30	+60	N.A.
131-102	M-4307-2	SA 508-CL2	Outlet Nozzle Safe End	10	+10	+30	+60	N.A.
128-501	M-6708-1	SA 508-CL2	Inlet Nozzle Extension	+20	+20	-10	+10	N.A.
128-501	M-6708-2	SA 508-CL2	Inlet Nozzle Extension	+20	+20	-10	+10	N.A.
128-501	M-6708-3	SA 508-CL2	Inlet Nozzle Extension	+20	+20	-20	+20	N.A.
128-501	M-6708-4	SA 508-CL2	Inlet Nozzle Extension	+20	+20	-20	+20	N.A.
120-301	M-4304-1	SA 508-CL2	Outlet Nozzle	-10	-10	-35**	-10**	N.A.
120-301	M-4304-2	SA 508-CL2	Outlet Nozzle	-10	-10	-35**	-10**	N.A.
131-101	M-6712-1	SA 508-CL1	Inlet Nozzle Safe End	-10	-10	+10	+45	N.A.
131-101	M-6712-2	SA 508-CL1	Inlet Nozzle Safe End	-10	-10	+10	+46	N.A.
131-101	M-6712-3	SA 508-CL1	Inlet Nozzle Safe End	-10	-10	+7	+50	N.A.
131-101	M-6712-4	SA 508-CL1	Inlet Nozzle Safe End	-10	-10	+7	+50	N.A.
120-101	M-6705-1	SA 508-CL2	Vessel Flange	70	70	+78	-28	N.A.
106-101	M-6706-1	SA 508-CL2	Closure Head Flange	70	70	88	54	N.A.

~~N.A. = Not Applicable (no minimum upper shelf requirement)~~
~~* = Lower bound curve values~~
~~** = Average of three test results~~
~~(a) = Determined per applicable ASME B&PV Code Sect. III, Subsection NB, Article NB-2331-(a-1,2,3).~~
~~(b) = 0° and 180° specimens had the same values~~

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PRESSURE/TEMPERATURE LIMITS (Continued)

The limitations imposed on the pressurizer heatup and cooldown rates and spray water temperature differential are provided to assure that the pressurizer is operated within the design criteria assumed for the fatigue analysis performed in accordance with the ASME Code requirements.

The OPERABILITY of two shutdown cooling suction line relief valves, one located in each shutdown cooling suction line, while maintaining the limits imposed on the RCS heatup and cooldown rates, ensures that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when one or more of the RCS cold legs are less than or equal to **°F during cooldown and **°F during heatup. Either one of the two SCS suction line relief valves provides relieving capability to protect the RCS from overpressurization when the transient is limited to either (1) the start of an idle RCP with the secondary water temperature of the steam generator less than or equal to 100°F above the RCS cold leg temperatures or (2) the inadvertent safety injection actuation with two HPSI pumps injecting into a water-solid RCS with full charging capacity and with letdown isolated. These events are the most limiting energy and mass addition transients, respectively, when the RCS is at low temperatures.

**

The limitations imposed on the RCS heatup and cooldown rates are provided to assure low temperature overpressure protection (LTOP) with the two shutdown cooling suction line relief valves operable. ~~Figure B 3/4.4-2 provides the limits of Appendix G to 10 CFR Part 50 for various heatup and cooldown rates.~~ At low temperatures with the relief valves aligned to the RCS, it is necessary to restrict heatup and cooldown rates to assure that the P/T limits are not exceeded. During worst case transients, RCS peak pressures can reach the relief valve setpoint, ** psig, plus accumulation. At temperatures greater than **°F during cooldown and **°F during heatup, the heatup and cooldown rate limitations assure the limits of Appendix G to 10 CFR 50 will not be exceeded with overpressure protection provided by the primary safety valves.

** **

3/4.4.9 STRUCTURAL INTEGRITY

The inservice inspection and testing programs for ASME Code Class 1, 2, and 3 components ensure that the structural integrity and operational readiness of these components will be maintained at an acceptable level throughout the life of the plant. These programs are in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50.55a(g) except where specific written relief has been granted by the Commission pursuant to 10 CFR 50.55a (g) (6) (i).

Components of the Reactor Coolant System were designed to provide access to permit inservice inspections in accordance with Section XI of the ASME Boiler and Pressure Vessel Code, 1974 Edition and Addenda through Summer 1975.

** See Applicant's SAR

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3/4.4.10 REACTOR COOLANT SYSTEM VENTS

Reactor Coolant System vents are provided to exhaust noncondensible gases and/or steam from the primary system that could inhibit natural circulation core cooling. The OPERABILITY of at least one Reactor Coolant System vent path from the reactor vessel head ensures the capability exists to perform this function. In addition, Branch Technical Position RSB 5-1 requires that a reactor vessel head vent path is OPERABLE in order to achieve a safe shutdown.

The valve redundancy of the Reactor Coolant System vent paths serves to minimize the probability of inadvertent or irreversible actuation while ensuring that a single failure of a vent valve, power supply, or control system does not prevent isolation of the vent path.

The function, capabilities, and testing requirements of the Reactor Coolant System vent systems are consistent with the requirements of Item II.B.1 of NUREG-0737.

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3/4.5.1 SAFETY INJECTION TANKS

The OPERABILITY of each of the Safety Injection System (SIS) safety injection tanks ensures that a sufficient volume of borated water will be immediately forced into the reactor core through each of the cold legs in the event the RCS pressure falls below the pressure of the safety injection tanks. This initial surge of water into the RCS provides the initial cooling mechanism during large RCS pipe ruptures.

The limits on safety injection tank volume, boron concentration, and pressure ensure that the safety injection tanks will adequately perform their function in the event of a LOCA in MODE 1, 2, 3, or 4.

A minimum of 25% narrow range corresponding to 1790 cubic feet and a maximum of 75% narrow range corresponding to 1927 cubic feet of borated water are used in the safety analysis as the volume in the SITs. To allow for instrument accuracy, 28% narrow range corresponding to 1802 cubic feet and 72% narrow range corresponding to 1914 cubic feet, are specified in the Technical Specification.

A minimum of 593 psig and a maximum pressure of 632 psig at normal operating cond.* are used in the safety analysis. To allow for instrument accuracy 600 psig minimum and 625 psig maximum are specified in the Technical Specification.

A boron concentration of ~~2000~~ 4000 ppm minimum and 4400 ppm maximum are used in the safety analysis.

The SIT isolation valves are not single failure proof; therefore, whenever the valves are open power shall be removed from these valves and the switch keylocked open. These precautions ensure that the SITs are available during a Limiting Fault.

The SIT nitrogen vent valves are not single failure proof against depressurizing the SITs by spurious opening. Therefore, power to the valves is removed while they are closed to ensure the safety analysis assumption of four pressurized SITs.

All of the SIT nitrogen vent valves are required to be operable so that, given a single failure, all four SITs may still be vented during post-LOCA long-term cooling. Venting the SITs provides for SIT depressurization capability which ensures the timely establishment of shutdown cooling entry conditions as assumed by the safety analysis for small break LOCAs.

The limits for operation with a safety injection tank inoperable for any reason except an isolation valve closed minimizes the time exposure of the plant to a LOCA event occurring concurrent with failure of an additional safety injection tank which may result in unacceptable peak cladding temperatures. If a closed isolation valve cannot be immediately opened, the full capability of one safety injection tank is not available and prompt action is required to place the reactor in a MODE where this capability is not required.

For MODES 3 and 4 operation with pressurizer pressure less than ~~1837~~ 1750 psia the Technical Specifications require a minimum of 57% wide range corresponding

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SAFETY INJECTION TANKS (Continued)

to 1361 cubic feet and a maximum of 75% narrow range corresponding to 1927 cubic feet of borated water per tank, when three safety injection tanks are operable and a minimum of 36% wide range corresponding to 908 cubic feet and a maximum of 75% narrow range corresponding to 1927 cubic feet per tank, when four safety injection tanks are operable at a minimum pressure of 235 psig and a maximum pressure of 625 psig. To allow for instrument inaccuracy, 60% wide range instrument corresponding to 1415 cubic feet, and 72% narrow range instrument corresponding to 1914 cubic feet, when three safety injection tanks are operable, and 39% wide range instrument corresponding to 962 cubic feet, and 72% narrow range instrument corresponding to 1914 cubic feet, when four SITs are operable, are specified in the Technical Specifications. To allow for instrument inaccuracy 254 psig is specified in the Technical Specifications.

3/4.5.2 and 3/4.5.3 ECCS SUBSYSTEMS

Insert 1

The OPERABILITY of two separate and independent ECCS subsystems ~~with the RCS temperature greater than or equal to 350°F~~ ensures that sufficient emergency core cooling capability will be available in the event of a LOCA assuming the loss of one subsystem through any single failure consideration. Either subsystem operating in conjunction with the safety injection tanks is capable of supplying sufficient core cooling to limit the peak cladding temperatures within acceptable limits for all postulated break sizes ranging from the double-ended break of the largest RCS cold leg pipe downward. In addition, each ECCS subsystem provides long-term core cooling capability in the recirculation mode during the accident recovery period.

Insert 2

~~With the RCS temperature below 350°F~~, one OPERABLE ECCS subsystem is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the limited core cooling requirements.

~~The trisodium phosphate dodecahydrate (TSP) stored in dissolving baskets located in the containment basement is provided to minimize the possibility of corrosion cracking of certain metal components during operation of the ECCS following a LOCA. The TSP provides this protection by dissolving in the sump water and causing its final pH to be raised to greater than or equal to 7.0.~~

The surveillance requirements provided to ensure OPERABILITY of each component ensure that at a minimum, the assumptions used in the safety analyses are met and that subsystem OPERABILITY is maintained. Surveillance requirements for throttle valve position stops and flow balance testing provide

Insert 1

for MODES 1 and 2 and for MODE 3 with the pressurizer pressure greater than or equal to 1750 psia

Insert 2

In MODE 3 with the pressurizer pressure less than 1750 psia and in MODE 4,

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ECCS SUBSYSTEMS (Continued)

assurance that proper ECCS flows will be maintained in the event of a LOCA.* Maintenance of proper flow resistance and pressure drop in the piping system to each injection point is necessary to: (1) prevent total pump flow from exceeding runout conditions when the system is in its minimum resistance configuration, (2) provide the proper flow split between injection points in accordance with the assumptions used in the ECCS-LOCA analyses, and (3) provide an acceptable level of total ECCS flow to all injection points equal to or above that assumed in the ECCS-LOCA analyses. ~~The requirement to dissolve a representative sample of TSP in a sample of RWT water provides assurance that the stored TSP will dissolve in borated water at the postulated post-LOCA temperatures.~~

The term "minimum bypass recirculation flow," as used in Specification 4.5.2e.3 and 4.5.2f., refers to that flow directed back to the RWT from the ECCS pumps for pump protection. Testing of the ECCS pumps under the condition of minimum bypass recirculation flow in Specification 4.5.2f. verifies that the performance of the ECCS pumps supports the safety analysis minimum RCS pressure assumption at zero delivery to the RCS.

3/4.5.4 REFUELING WATER TANK

The OPERABILITY of the refueling water tank (RWT) as part of the ECCS ensures that a sufficient supply of borated water is available for injection by the ECCS in the event of a LOCA. The limits on RWT minimum volume and boron concentration ensure that (1) sufficient water plus 10% margin is available to permit 20 minutes of engineered safety features pump operation, and (2) the reactor will remain subcritical in the cold condition following mixing of the RWT and the RCS water volumes with all control rods inserted except for the most reactive control assembly. These assumptions are consistent with the LOCA analyses.

---

*The following test conditions, which apply during flow balance tests, ensure that the ECCS subsystems are adequately tested.

3.2

1. The pressurizer pressure is at ~~atmospheric pressure~~ 15 psia.
2. The miniflow bypass recirculation lines are aligned for injection.
3. For LPSI system, (add/subtract) ~~[illegible]~~ gpm (to/from) the ~~[illegible] gpm~~ 2450 requirement for every foot by which the difference of RWT water level above the RWT RAS setpoint level (exceeds/is less than) the difference of RCS water level above the cold leg centerline.

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REFUELING WATER TANK (Continued)

The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

The limits on contained water volume and boron concentration of the RWT also ensure a pH value of between ~~7.0 and 8.5~~ ** for the solution recirculated within containment after a LOCA. ~~This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components.~~

The limit on the RWT solution temperature ensures that the assumptions used in the LOCA analyses remain valid.

** See Applicant's SAR

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BASES

3/4.6.1 PRIMARY CONTAINMENT

See Applicant's SAR

~~3/4.6.1.1 CONTAINMENT INTEGRITY~~

Primary CONTAINMENT INTEGRITY ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the safety analyses. This restriction, in conjunction with the leakage rate limitation, will limit the site boundary radiation doses to within the limits of 10 CFR Part 100 during accident conditions.

3/4.6.1.2 CONTAINMENT LEAKAGE

The limitations on containment leakage rates ensure that the total containment leakage volume will not exceed the value assumed in the safety analyses at the peak accident pressure, P_a. As an added conservatism, the measured overall integrated leakage rate is further limited to less than or equal to 0.75 L_a or less than or equal to 0.75 L_t, as applicable during performance of the periodic tests to account for possible degradation of the containment leakage barriers between leakage tests.

The surveillance testing for measuring leakage rates are consistent with the requirements of Appendix J of 10 CFR Part 50.

3/4.6.1.3 CONTAINMENT AIR LOCKS

The limitations on closure and leak rate for the containment air locks are required to meet the restrictions on CONTAINMENT INTEGRITY and containment leak rate. Surveillance testing of the air lock seals provides assurance that the overall air lock leakage will not become excessive due to seal damage during the intervals between air lock leakage tests.

3/4.6.1.7 CONTAINMENT VENTILATION SYSTEM

The 42-inch containment purge supply and exhaust isolation valves are required to be closed during plant operation since these valves have not been demonstrated capable of closing during a LOCA or steam line break accident. Maintaining these valves closed during plant operations ensures that excessive quantities of radioactive materials will not be released via the containment purge system. To provide assurance that the 42-inch valves cannot be inadvertently opened, they are sealed closed in accordance with Standard Review Plan 6.2.4 which includes mechanical devices to seal or lock the valve closed, or prevent power from being supplied to the valve operator.

The use of the containment purge lines is restricted to the 8-inch purge supply and exhaust isolation valves since, unlike the 42-inch valves, the 8-inch valves will close during a LOCA or steam line break accident and therefore the site boundary dose guidelines of 10 CFR Part 100 would not be exceeded in the event of an accident during purging operations.

Leakage integrity tests with a maximum allowable leakage rate for purge supply and exhaust isolation valves will provide early indication of resilient material seal degradation and will allow the opportunity for repair before gross leakage failure develops. The 0.60 L_a leakage limit shall not be exceeded when the leakage rates determined by the leakage integrity tests of these valves are added to the previously determined total for all valves and ~~penetrations subject to Type B and C tests.~~

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

3/4.6.2.1 CONTAINMENT SPRAY SYSTEM

The OPERABILITY of the containment spray system ensures that containment depressurization and cooling capability will be available in the event of a LOCA. The pressure reduction and resultant lower containment leakage rate are consistent with the assumptions used in the safety analyses.

The containment spray system and the containment cooling system are redundant to each other in providing post-accident cooling of the containment atmosphere. However, the containment spray system also provides a mechanism for removing iodine from the containment atmosphere and therefore the time requirements for restoring an inoperable spray system to OPERABLE status have been maintained consistent with that assigned other inoperable ESF equipment.

3/4.6.2.2 ~~IODINE REMOVAL~~ SYSTEM SPRAY ADDITIVE

The OPERABILITY of the ~~iodine removal~~ system ensures that sufficient N_2H_4 is added to the containment spray in the event of a LOCA. The limits on N_2H_4 volume and concentration ensure adequate chemical available to remove iodine from the containment atmosphere following a LOCA.

spray additive

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← Insert 1

3/4.6.~~3~~4 CONTAINMENT ISOLATION VALVES

The OPERABILITY of the containment ~~automatic~~ isolation valves ensures that the containment atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the containment atmosphere or pressurization of the containment and is consistent with the requirements of GDC 54 through GDC 57 of Appendix A to 10 CFR Part 50. Containment isolation within the time limits specified for those isolation valves designed to close automatically ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a LOCA.

~~The only valves in the Table 6.2.4-1 of the PVNGS FSAR that are not required to be listed in Table 3.6-1 are the following: main steam safety valves and main steam atmospheric dump valves. The main steam safety valves and the atmospheric dump valves have very high pressure setpoints to actuate and are covered by Specifications 3/4.7.1.1 and 3/4.7.1.6, respectively.~~

3/4.6.~~4~~5 COMBUSTIBLE GAS CONTROL

See Applicant's SAR.

~~The OPERABILITY of the equipment and systems required for the detection and control of hydrogen gas ensures that this equipment will be available to maintain the hydrogen concentration within containment below its flammable limit during post-LOCA conditions. Either recombiner unit (or the purge system) is capable of controlling the expected hydrogen generation associated with (1) zirconium-water reactions, (2) radiolytic decomposition of water and (3) corrosion of metals within containment. These hydrogen control systems are consistent with the recommendations of Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a LOCA," March 1971.~~

~~There is a hydrogen monitor and a oxygen monitor in the Post Accident Sampling System (PASS). There is a line from the two containment hydrogen monitors to the PASS which allows the licensee to monitor hydrogen concentration in the containment following an accident with the PASS. An OPERABLE hydrogen monitor in the PASS allows the licensee to enter Modes 1 and 2 with one containment hydrogen monitor inoperable. For the hydrogen monitor in PASS to be OPERABLE, the valves in the piping to the monitor must be OPERABLE.~~

← Insert 2

Insert 1

3/4.6.2.3 CONTAINMENT COOLING SYSTEM

See Applicant's SAR.

(OPTIONAL)

3/4.6.3 IODINE CLEANUP SYSTEM

See Applicant's SAR.

Insert 2

3/4.6.6 PENETRATION ROOM EXHAUST AIR CLEANUP SYSTEM (OPTIONAL)

See Applicant's SAR.

3/4.6.7 VACUUM RELIEF VALVES (OPTIONAL

See Applicant's SAR.

3/4.6.8 SECONDARY CONTAINMENT (DUAL TYPE CONTAINMENT, OPTIONAL)

See Applicant's SAR.

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3/4.7.1 TURBINE CYCLE

3/4.7.1.1 SAFETY VALVES

1380 1255

pressure

The OPERABILITY of the main steam safety valves (MSSVs) limit secondary system pressure to within 110% (~~[illegible]~~ psia) of the design pressure (~~[illegible]~~ psia) during the most severe anticipated operational transient. For design purposes the valves are sized to pass a minimum of 102% of the RATED THERMAL POWER at 102% of design ~~power~~. The adequacy of this relieving capacity is demonstrated by maintaining the Reactor Coolant System pressure below NRC acceptance criteria (120% of design pressure for large feedwater line breaks, CEA ejection and 110% of design pressure for all overpressurization events).

The specified valve lift settings and relieving capacities are in accordance with the requirements of Section III of the ASME Boiler and Pressure Vessel Code, ~~1974 Edition including the Summer 1975 Addenda~~. ** The total relieving capacity for all twenty MSSVs at 110% of system design pressure (adjusted for 50 psi pressure drop to valves inlet) is ~~[illegible]~~ 18.66 x 10^6 lbm/hr. This capacity is less than the total rated capacity of 19 x 10^6 lbm/hr given in FSAR T. 10.3-2 as the MSSVs are operating at an inlet pressure below rated conditions. At these same secondary pressure conditions, the total steam flow at 102% (2% uncertainty) of 3817 MWt (RATED THERMAL POWER plus 17 MWt pump heat input) is 17.83 x 10^6 lbm/hr. The ratio of this total steam flow to the total capacity is ~~[illegible]~~ 104.6%.

STARTUP and/or POWER OPERATION is allowable with MSSVs inoperable if the maximum allowable power level is reduced to a value equal to the product of the ratio of the number of MSSVs available per steam generator to the total number of MSSVs per steam generator with the ratio of the total steam flow to available relieving capacity.

$$\text{Allowable Power Level} = \left(\frac{10-N}{10}\right) \times \frac{104.6}{\cancel{[illegible]}}$$

The ceiling on the variable over power reactor trip is also reduced to an amount over the allowable power level equal to the BAND given for this trip in Table 2.2-1.

SP = Allowable Power Level + 9.8

where:

SP = reduced reactor trip setpoint in percent of RATED THERMAL POWER. This is a ratio of the available relieving capacity over the total steam flow at rated power.

** See Applicant's SAR for appropriate Edition of ASME Code.

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SAFETY VALVES (continued)

10 = total number of secondary safety valves for one steam generator.

N = number of inoperable main steam safety valves on the steam generator with the greater number of inoperable valves.

104.6 ~~[illegible]~~ = ratio of main steam safety valve relieving capacity of 110% steam generator design pressure to calculated steam flow rate at 100% plant power + 2% uncertainty (see above text)

9.8 = BAND between the maximum thermal power and the variable over-power trip setpoint ceiling

3/4.7.1.2 EMERGENCY ~~AUXILIARY~~ FEEDWATER SYSTEM

R See Applicant's SAR.

~~The OPERABILITY of the auxiliary feedwater system ensures that the Reactor~~ Coolant System can be cooled down to less than 350°F from normal operating conditions in the event of a total loss-of-offsite power.

Each electric-driven auxiliary feedwater pump is capable of delivering a minimum feedwater flow of 750 gpm at a pressure of 1270 psia to the entrance of the steam generators. The steam-driven auxiliary feedwater pump is capable of delivering a minimum feedwater flow of 750 gpm at a pressure of 1270 psia to the entrance of the steam generators. This capacity is sufficient to ensure that adequate feedwater flow is available to remove decay heat and reduce the Reactor Coolant System temperature to less than 350°F when the shutdown cooling system may be placed into operation.

3/4.7.1.3 CONDENSATE STORAGE TANK

The OPERABILITY of the condensate storage tank ensures that a minimum water volume of 300,000 gallons is available to maintain the Reactor Coolant System at HOT STANDBY for 6 hours followed by an orderly cooldown to the shutdown cooling entry (350°F) temperature with concurrent total loss-of-site power, and also ensures that sufficient water is available to maintain the RCS at HOT STANDBY conditions for 8 hours with steam discharge to atmosphere concurrent with total loss-of-offsite power. The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

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3/4.7.1.4 ACTIVITY

The limitations on secondary system specific activity ensure that the resultant offsite radiation dose will be limited to a small fraction of 10 CFR Part 100 limits in the event of a steam line rupture. This dose also includes the effects of a coincident 1 gpm primary-to-secondary tube leak in the steam generator of the affected steam line and a concurrent loss-of-offsite electrical power. These values are consistent with the assumptions used in the safety analyses.

3/4.7.1.5 MAIN STEAM LINE ISOLATION VALVES

The OPERABILITY of the main steam line isolation valves ensures that no more than one steam generator will blow down in the event of a steam line rupture. This restriction is required to (1) minimize the positive reactivity effects of the Reactor Coolant System cooldown associated with the blowdown, and (2) limit the pressure rise within containment in the event the steam line rupture occurs within containment. The OPERABILITY of the main steam isolation valves within the closure times of the surveillance requirements are consistent with the assumptions used in the safety analyses.

3/4.7.1.6 ATMOSPHERIC DUMP VALVES

~~The limitation on maintaining the nitrogen accumulator at a pressure > 400 psig is to ensure that a sufficient volume of nitrogen is in the accumulator to operate the associated ADV which holds the plant at hot standby while dissipating core decay heat or which allows a flow of sufficient steam to maintain a controlled reactor cooldown rate. A pressure of 400 psig retains sufficient nitrogen volume for 4 hours of operation at hot standby plus 6.5 hours of operation to reach cold shutdown under natural circulation conditions in the event of failure of the normal control air system.~~

Insert

3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

The limitation on steam generator pressure and temperature ensures that the pressure induced stresses in the steam generators do not exceed the maximum allowable fracture toughness stress limits. The limitations to [illegible]°F and ** [illegible] psig are based on a steam generator RT_{NDT} of [illegible]°F and are sufficient to prevent brittle fracture. ** **

3/4.7.3 ~~ESSENTIAL~~ COMPONENT COOLING WATER SYSTEM

See Applicant's SAR.

~~The OPERABILITY of the essential cooling water system ensures that sufficient cooling capacity is available for continued operation of safety related equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the safety analyses.~~

** See Applicant's SAR

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Insert

~~3/4.7.1.6 ATMOSPHERIC DUMP VALVES~~ A

One atmospheric dump valve is required to hold the plant at HOT STANDBY and to conduct a controlled cooldown to shutdown cooling entry conditions, with the condensor unavailable, one steam generator unavailable for heat removal, and a single failure of one of the atmospheric dump valves on the available steam generator. The block valves are required to isolate an atmospheric dump valve which cannot be tightly closed.

SERVICE WATER

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See Applicant's SAR.

3/4.7.4 ~~ESSENTIAL SPRAY POND~~ SYSTEM

~~The OPERABILITY of the essential spray pond system ensures that sufficient cooling capacity is available for continued operation of equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the safety analyses.~~

3/4.7.5 ULTIMATE HEAT SINK

~~The limitations on the ultimate heat sink level and temperature ensure that~~ sufficient cooling capacity is available to either (1) provide normal cooldown of the facility, or (2) to mitigate the effects of accident conditions within acceptable limits.

The limitations on minimum water level and maximum temperature are based on providing a 27-day cooling water supply to safety-related equipment without exceeding their design basis temperature and is consistent with the intent of the recommendations of Regulatory Guide 1.27, "Ultimate Heat Sink for Nuclear ~~Plants," March 1974.~~

FLOOD PROTECTION

3/4.7.6 ~~ESSENTIAL CHILLED WATER SYSTEM~~

~~The OPERABILITY of the essential chilled water system ensures that suffi~~cient cooling capacity is available for continued operation of equipment and control room habitability during accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the safety analyses.

The Essential Chilled Water System (ECWS), in conjunction with respective emergency HVAC units, is required in accordance with Specification Definition 1.18 to provide heat removal in maintaining the various Engineered Safety Features (ESFs) room space design temperatures below the associated equipment qualification limits for the range of Design Basis Accident conditions. The normal HVAC system is redundant to the emergency HVAC system in maintaining the space design conditions of required safety systems during normal operating conditions and Design Basis Accident Conditions not involving seismic events or loss of offsite power. A seven (7) day Action requirement is for a single ECWS out of service, based on the high reliability of offsite power and availability of the normal HVAC system. The normal HVAC system contains two 100% redundant chillers. Action requirements are provided to ensure operability of the vital bus inverters and emergency battery chargers, by verifying within one hour that the normal HVAC system is providing space cooling to the vital power distribution rooms. The Action requirement is provided to establish within 8 hours operability of the safe shutdown systems which do not depend on the inoperable ECWS. The 8 hour period provides a reasonable time in which to establish operability of this complement of key safety systems. This requirement ensures that a functional train of safe shutdown equipment is available to put the plant in a safe, stable condition for the most probable abnormal operational occurences. An Action requirement of 24 hours is provided to establish operability of the remaining required safety ~~systems which do not depend on the inoperable ECWS.~~

See Applicant's SAR.

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EMERGENCY AIR CLEANUP

3/4.7.7 CONTROL ROOM ~~ESSENTIAL FILTRATION~~ SYSTEM

See Applicant's SAR

~~The OPERABILITY of the control room essential filtration system ensures that the control room will remain habitable for operations personnel during and following all credible accident conditions. The OPERABILITY of this system in conjunction with control room design provisions is based on limiting the radiation exposure to personnel occupying the control room to 5 rem or less whole body, or its equivalent. This limitation is consistent with the requirements of General Design Criterion 19 of Appendix A, 10 CFR Part 50.~~

ECCS

3/4.7.8 ~~ESF~~ PUMP ROOM AIR EXHAUST CLEANUP SYSTEM

See Applicant's SAR

~~The OPERABILITY of the ESF pump room air exhaust cleanup system ensures that radioactive materials leaking from the ECCS equipment within the pump room following a LOCA are filtered prior to reaching the environment. The operation of this system and the resultant effect on offsite dosage calculations was assumed in the safety analyses.~~

3/4.7.9 SNUBBERS

All snubbers are required OPERABLE to ensure that the structural integrity of the reactor coolant system and all other safety-related systems is maintained during and following a seismic or other event initiating dynamic loads. Snubbers excluded from this inspection program are those installed on nonsafety-related systems and then only if their failure or failure of the system on which they are installed, would have no adverse effect on any safety-related system.

Snubbers are classified and grouped by design and manufacturer but not by size. For example, mechanical snubbers utilizing the same design features of the 2-kip, 10-kip, and 100-kip capacity manufactured by Company "A" are of the same type. The same design mechanical snubbers manufactured by Company "B" for the purposes of this Technical Specification would be of a different type, as would hydraulic snubbers from either manufacturer.

A list of individual snubbers with detailed information of snubber location and size and of system affected shall be available at the plant in accordance with Section 50.71(c) of 10 CFR Part 50. ~~The accessibility of each snubber shall be determined and approved by the Plant Review Board. The determination shall be based upon the existing radiation levels and the expected time to perform a visual inspection in each snubber location as well as other factors associated with accessibility during plant operations (e.g., temperature, atmosphere, location, etc.), and the recommendations of Regulatory Guides 8.8 and 8.10.~~ The addition or deletion of any hydraulic or mechanical snubber shall be made in accordance with Section 50.59 of 10 CFR Part 50.

The visual inspection frequency is based upon maintaining a constant level of snubber protection. Therefore, the required inspection interval varies inversely with the observed snubber failures and is determined by the number

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SNUBBERS (Continued)

of inoperable snubbers found during an inspection. ~~In order to establish the inspection frequency for each type of snubber, it was assumed that the frequency of failures and initiating events is constant with time and that the failure of any snubber could cause that system to be unprotected and to result in failure during an assumed initiating event.~~ Inspections performed before that interval has elapsed may be used as a new reference point to determine the next inspection. However, the results of such early inspections performed before the original required time interval has elapsed (nominal time less 25%) may not be used to lengthen the required inspection interval. Any inspection whose results require a shorter inspection interval will override the previous schedule.

The acceptance criteria are to be used in the visual inspection to determine OPERABILITY of the snubbers.

To provide assurance of snubber functional reliability one of three functional testing methods are used with the stated acceptance criteria:

1. Functionally test 10% of a type of snubber with an additional 10% tested for each functional testing failure, or

2. Functionally test a sample size and determine sample acceptance or rejection using Figure 4.7-1, or

3. Functionally test a representative sample size and determine sample acceptance or rejection using the stated equation.

Figure 4.7-1 was developed using "Wald's Sequential Probability Ratio Plan" as described in "Quality Control and Industrial Statistics" by Acheson J. Duncan.

Permanent or other exemptions from the surveillance program for individual snubbers may be granted by the Commission if a justifiable basis for exemption is presented and, if applicable, snubber life destructive testing was performed to qualify the snubbers for the applicable design conditions at either the completion of their fabrication or at a subsequent date. Snubbers so exempted shall be listed in the list of individual snubbers indicating the extent of the exemptions.

The service life of a snubber is established via manufacturer input and information through consideration of the snubber service conditions and associated installation and maintenance records (newly installed snubber, seal replaced, spring replaced, in high radiation area, in high temperature area, etc.). The requirement to monitor the snubber service life is included to ensure that the snubbers periodically undergo a performance evaluation in view of their age and operating conditions. These records will provide statistical bases for future consideration of snubber service life.

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3/4.7.10 SEALED SOURCE CONTAMINATION

See Applicant's SAR.

~~The limitations on removable contamination for sources requiring leak testing, including alpha emitters, is based on 10 CFR 70.39(c) limits for plutonium. This limitation will ensure that leakage from byproduct, source, and special nuclear material sources will not exceed allowable intake values.~~

~~Sealed sources are classified into three groups according to their use, with surveillance requirements commensurate with the probability of damage to a source in that group. Those sources which are frequently handled are required to be tested more often than those which are not. Sealed sources which are continuously enclosed within a shielded mechanism (i.e. sealed sources within radiation monitoring or boron measuring devices) are considered to be stored and need not be tested unless they are removed from the shield mechanism.~~

3/4.7.11 FIRE SUPPRESSION SYSTEMS

See Applicant's SAR.

~~The OPERABILITY of the fire suppression systems ensures that adequate fire suppression capability is available to confine and extinguish fires occurring in any portion of the facility where safety-related equipment is located. The fire suppression system consists of the water system, spray and/or sprinklers, CO_2, Halon, fire hose stations, and yard fire hydrants. The collective capability of the fire suppression systems is adequate to minimize potential damage to safety-related equipment and is a major element in the facility fire protection program.~~

~~In the event that portions of the fire suppression systems are inoperable, alternate backup fire fighting equipment is required to be made available in the affected area(s) until the inoperable equipment is restored to service. When the inoperable fire fighting equipment is intended for use as a backup means of fire suppression, a longer period of time is allowed to provide an alternate means of fire fighting than if the inoperable equipment is the primary means of fire suppression.~~

~~The surveillance requirements provide assurance that the minimum OPERABILITY requirements of the fire suppression systems are met. An allowance is made for ensuring a sufficient volume of CO_2/Halon in the CO_2/Halon storage tank by verifying either the weight or the level of the tank.~~

~~In the event the fire suppression water system becomes inoperable, immediate corrective measures must be taken since this system provides the major fire suppression capability of the plant. The requirement for a 24-hour report to the Commission provides for prompt evaluation of the acceptability of the corrective measures to provide adequate fire suppression capability for the continued protection of the nuclear plant.~~

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~~3/4.7.12 FIRE-RATED ASSEMBLIES~~ See Applicant's SAR

~~The OPERABILITY of the fire barriers and barrier penetrations ensure that fire damage will be limited. These design features minimize the possibility of a single fire involving more than one fire area prior to detection and extinguishment. The fire barriers, fire barrier penetrations for conduits, cable trays and piping, fire dampers, and fire doors are periodically inspected and functionally tested to verify their OPERABILITY.~~

3/4.7.~~13~~14 SHUTDOWN COOLING SYSTEM

The OPERABILITY of two separate and independent shutdown cooling subsystems ensures that the capability of initiating shutdown cooling in the event of an accident exists even assuming the most limiting single failure occurs. The safety analysis assumes that shutdown cooling can be initiated when conditions permit.

The limits of operation with one shutdown cooling inoperable for any reason minimize the time exposure of the plant to an accident event occurring concurrent with the failure of a component on the other shutdown cooling subsystem.

~~3/4.7.14 CONTROL ROOM AIR TEMPERATURE~~

~~Maintaining the control room air temperature less than or equal to 80°F ensures that (1) the ambient air temperature does not exceed the allowable air temperature for continuous duty rating for the equipment and instrumentation in the control room and (2) the control room will remain habitable for operations personnel during plant operation. The 30 days to return the control room air temperature to less than or equal to 80°F in the Action Statement is consistent with the equipment qualification program for the control room.~~

3/4.7.13 AREA TEMPERATURE MONITORING

See Applicant's SAR

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~~3/4.8.1, 3/4.8.2 and 3/4.8.3 A.C. SOURCES, D.C SOURCES and ONSITE POWER DISTRIBUTION SYSTEMS~~

~~The OPERABILITY of the A.C. and D.C. power sources and associated~~ distribution systems during operation ensures that sufficient power will be available to supply the safety-related equipment required for (1) the safe shutdown of the facility and (2) the mitigation and control of accident conditions within the facility. The minimum specified independent and redundant A.C. and D.C. power sources and distribution systems satisfy the requirements of General Design Criterion 17 of Appendix "A" to 10 CFR 50.

The ACTION requirements specified for the levels of degradation of the power sources provide restriction upon continued facility operation commensurate with the level of degradation. The OPERABILITY of the power sources are consistent with the initial condition assumptions of the safety analyses and are based upon maintaining at least one redundant set of onsite A.C. and D.C. power sources and associated distribution systems OPERABLE during accident conditions coincident with an assumed loss-of-offsite power and single failure of the other onsite A.C. source. The A.C. and D.C. source allowable out-of-service times are based on Regulatory Guide 1.93, "Availability of Electrical Power Sources," December 1974. When one diesel generator is inoperable, there is an additional ACTION requirement to verify that all required systems, subsystems, trains, components, and devices that depend on the remaining OPERABLE diesel generator as a source of emergency power are also OPERABLE, and that the steam-driven auxiliary feedwater pump is OPERABLE. This requirement is intended to provide assurance that a loss-of-offsite power event will not result in a complete loss of safety function of critical systems during the period one of the diesel generators is inoperable. The term verify as used in this context means to administratively check by examining logs or other information to determine if certain components are out-of-service for maintenance or other reasons. It does not mean to perform the surveillance requirements needed to demonstrate the OPERABILITY of the component.

Insert

The required steady state frequency for the emergency diesels is 60 + 1.2/ -0.3 Hz to be consistent with the safety analysis to provide adequate safety injection flow.

The OPERABILITY of the minimum specified A.C. and D.C. power sources and associated distribution systems during shutdown and refueling ensures that (1) the facility can be maintained in the shutdown or refueling condition for extended time periods and (2) sufficient instrumentation and control capability is available for monitoring and maintaining the unit status.

The surveillance requirements for demonstrating the OPERABILITY of the diesel generators are in accordance with the recommendations of Regulatory Guides 1.9 "Selection of Diesel Generator Set Capacity for Standby Power Supplies," March 10, 1971, and 1.108 "Periodic Testing of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants," Revision 1, August 1977, and 1.137, "Fuel Oil Systems for Standby Diesel Generators," Revision 1, October 1979.

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3/4.8.1 A.C. SOURCES

See Applicant's SAR.

3/4.8.2 D.C. SOURCES

See Applicant's SAR.

3/4.8.3 ONSITE POWER DISTRIBUTION

See Applicant's SAR.

3/4.8.4 ELECTRICAL EQUIPMENT PROTECTIVE DEVICES

See Applicant's SAR.

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BASES

3/4.9.1 BORON CONCENTRATION

The limitations on reactivity conditions during REFUELING ensure that: (1) the reactor will remain subcritical during CORE ALTERATIONS, and (2) a uniform boron concentration is maintained for reactivity control in the water volume having direct access to the reactor vessel. These limitations are consistent with the initial conditions assumed for the boron dilution incident in the safety analyses. The value of 0.95 or less for K_{eff} includes a 1% delta k/k conservative allowance for uncertainties. Similarly, the boron concentration value of 2150 ppm or greater also includes a conservative uncertainty allowance of 50 ppm boron.

3/4.9.2 INSTRUMENTATION

The OPERABILITY of the source range neutron flux monitors ensures that redundant monitoring capability is available to detect changes in the reactivity condition of the core.

3/4.9.3 DECAY TIME

The minimum requirement for reactor subcriticality prior to movement of irradiated fuel assemblies in the reactor pressure vessel ensures that sufficient time has elapsed to allow the radioactive decay of the short lived fission products. This decay time is consistent with the assumptions used in the safety analyses.

3/4.9.4 CONTAINMENT ~~BUILDING~~ PENETRATIONS

R See Applicant's SAR.

~~The requirements on containment penetration closure and OPERABILITY ensure that a release of radioactive material within containment will be restricted from leakage to the environment. The OPERABILITY and closure restrictions are sufficient to restrict radioactive material release from a fuel element rupture based upon the lack of containment pressurization potential while in the REFUELING MODE.~~

3/4.9.5 COMMUNICATIONS

R See Applicant's SAR.

~~The requirement for communications capability ensures that refueling station personnel can be promptly informed of significant changes in the facility status or core reactivity condition during CORE ALTERATIONS.~~

MANIPULATOR CRANE

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3/4.9.6 ~~REFUELING MACHINE~~

R See Applicant's SAR.

~~The OPERABILITY requirements for the refueling machine ensure that: (1) the machine will be used for movement of fuel assemblies, (2) the machine has sufficient load capacity to lift a fuel assembly, and (3) the core internals and pressure vessel are protected from excessive lifting force in the event they are inadvertently engaged during lifting operations.~~

3/4.9.7 CRANE TRAVEL - SPENT FUEL STORAGE POOL BUILDING

R See Applicants' SA

~~The restriction on movement of loads in excess of the nominal weight of a fuel assembly, CEA and associated handling tool over other fuel assemblies in the storage pool ensures that in the event this load is dropped (1) the activity release will be limited to that contained in a single fuel assembly, and (2) any possible distortion of fuel in the storage racks will not result in a critical array. This assumption is consistent with the activity release assumed in the safety analyses.~~

3/4.9.8 SHUTDOWN COOLING AND COOLANT CIRCULATION

The requirement that at least one shutdown cooling loop be in operation, and circulating reactor coolant at a flow rate equal to or greater than 4000 gpm ensures that (1) sufficient cooling capacity is available to remove decay heat and maintain the water in the reactor pressure vessel below 135°F as required during the REFUELING MODE, (2) sufficient coolant circulation is maintained through the reactor core to minimize the effects of a boron dilution incident and prevent boron stratification, and (3) the ΔT across the core will be maintained at less than 75°F during the REFUELING MODE. The required flowrate of $\geq$ 4000 gpm ensures that ~~XX~~ hours after reactor shutdown sufficient cooling capacity is available to remove decay heat and maintain the water in the reactor pressure vessel below 135°F as required during the REFUELING MODE. ~~This assumes a shutdown cooling heat exchanger cooling water flowrate of 14000 gpm, a cooling water inlet temperature of < 105°F at > 27 1/2 hours after reactor shutdown, and the decay heat curve of CESSAR F Figure 6.3.-3 and reactor operation for two years at 4000 MWt.~~ **

Without a shutdown cooling train in operation steam may be generated; therefore, the containment should be sealed off to prevent escape of any radioactivity, and any operations that would cause an increase in decay heat should be secured.

The requirement to have two shutdown cooling loops OPERABLE when there is less than 23 feet of water above the reactor pressure vessel flange, ensures that a single failure of the operating shutdown cooling loop will not result in a complete loss of decay heat removal capability. With the reactor vessel head removed and 23 feet of water above the reactor pressure vessel flange, a large heat sink is available for core cooling, thus in the event of a failure of the operating shutdown cooling loop, adequate time is provided to initiate emergency procedures to cool the core.

** See Applicant's SAR

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BASES

~~A shutdown cooling loop may be removed from operation for up to 1 hour per 8-hour period during surveillance testing of ECCS pumps. This is necessary to meet Surveillance 4.5.2, flow testing of the HPSI pumps without other pumps running, and 4.5.3.5, testing of the containment spray pumps and LPSI pumps during surveillance of the remote shutdown system.~~

3/4.9.9 CONTAINMENT PURGE VALVE ISOLATION SYSTEM

See Applicant's SAR.

~~The OPERABILITY of this system ensures that the containment purge valves will be automatically isolated upon detection of high radiation levels within the containment. The OPERABILITY of this system is required to restrict the release of radioactive material from the containment atmosphere to the environment.~~

3/4.9.10 and 3/4.9.11 WATER LEVEL - REACTOR VESSEL and STORAGE POOL

The restrictions on minimum water level ensure that sufficient water depth (at least 23 feet above the top of the spent fuel) is available to remove a nominal 99% of the assumed 10% iodine gap activity released from the rupture of an irradiated fuel assembly for a maximum fuel rod pressurization of 1200 psig. The minimum water depth is consistent with the assumptions of the safety analysis.

3/4.9.12 ~~FUEL BUILDING ESSENTIAL VENTILATION~~ STORAGE POOL AIR CLEANUP SYSTEM

See Applicant's SAR

~~The limitations on the fuel building essential ventilation system ensure that all radioactive material released from an irradiated fuel assembly will be filtered through the HEPA filters and charcoal adsorber prior to discharge to the atmosphere. The OPERABILITY of this system and the resulting iodine removal capacity are consistent with the assumptions of the safety analyses.~~

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BASES

3/4.10.1 SHUTDOWN MARGIN

This special test exception provides that a minimum amount of CEA worth is immediately available for reactivity control when tests are performed for CEAs worth measurement. This special test exception is required to permit the periodic verification of the actual versus predicted core reactivity condition occurring as a result of fuel burnup or fuel cycling operations. Although testing will be initiated from MODE 2, temporary entry into MODE 3 is necessary during some CEA worth measurements. A reasonable recovery time is available for return to MODE 2 in order to continue PHYSICS TESTING.

3/4.10.2 MODERATOR TEMPERATURE COEFFICIENT, GROUP HEIGHT, INSERTION, AND POWER DISTRIBUTION LIMITS

This special test exception permits individual CEAs to be positioned outside of their normal group heights and insertion limits during the performance of such PHYSICS TESTS as those required to (1) measure CEA worth, (2) determine the reactor stability index and damping factor under xenon oscillation conditions, (3) determine power distributions for non-normal CEA configurations, (4) measure rod shadowing factors, and (5) measure temperature and power coefficients. ~~A~~ This special test exception permits MTC to exceed limits in Specification 3.1.1.3 during performance of PHYSICS TESTS.

3/4.10.3 REACTOR COOLANT LOOPS

This special test exception permits reactor criticality with less than four reactor coolant pumps in operation and is required to perform certain STARTUP and PHYSICS TESTS while at low THERMAL POWER levels.

3/4.10.4 CEA POSITION, REGULATING CEA INSERTION LIMITS AND REACTOR COOLANT COLD LEG TEMPERATURE

This special test exception permits the CEAs to be positioned beyond the insertion limits and reactor coolant cold leg temperature to be outside limits during PHYSICS TESTS required to determine the isothermal temperature coefficient and power coefficient.

3/4.10.5 MINIMUM TEMPERATURE AND PRESSURE FOR CRITICALITY

This special test exception permits reactor criticality at low THERMAL POWER levels with T_{cold} below the minimum critical temperature and pressure during PHYSICS TESTS which are required to verify the low temperature physics predictions and to ensure the adequacy of design codes for reduced temperature conditions. The Low Power Physics Testing Program at low temperature (~~[illegible]~~°F) 320°F and a pressure of 500 psia is used to perform the following tests:

1. Biological shielding survey test
2. Isothermal temperature coefficient tests
3. CEA group tests
4. Boron worth tests
5. Critical configuration boron concentration

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3/4.10.6 SAFETY INJECTION TANKS

This special test exception permits testing the low pressure safety injection system check valves. The pressure in the injection header must be reduced below the head of the low pressure injection pump in order to get flow through the check valves. The safety injection tank (SIT) isolation valve must be closed in order to accomplish this. The SIT isolation valve is still capable of automatic operation in the event of an SIAS; therefore, system capability should not be affected.

~~3/4.10.7 SPENT FUEL POOL LEVEL~~

~~This special test exception permits loading of the initial core with the spent fuel pool dry.~~

3/4.10.~~8~~7 SAFETY INJECTION TANK PRESSURE

This special test exception allows the performance of PHYSICS TESTS at low pressure/low temperature (600 psig, 320°F) conditions which are required to verify the low temperature physics predictions and to ensure the adequacy of design codes for reduced temperature conditions.

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See Applicant's SAR

3/4.11.1 SECONDARY SYSTEM LIQUID WASTE DISCHARGE TO ONSITE EVAPORATION PONDS

~~3/4.11.1.1 CONCENTRATION~~

This specification is provided to ensure that at any time during the life of the nuclear station, the annual total body dose due to ground contamination of an UNRESTRICTED AREA, arising from transportation and deposition by wind of the accumulated activity discharged to the pond from the secondary system of the plant (if the pond gets dried up) on the UNRESTRICTED AREA, is within the guidelines of 10 CFR Part 20 for the above-mentioned postulated event.

Restricting the concentrations of the secondary liquid wastes discharged to the onsite evaporation ponds will restrict the quantity of radioactive material that can get accumulated in the ponds. This, in turn, provides assurance that in the event of an uncontrolled release of the pond's contents to an UNRESTRICTED AREA, the resulting total body annual exposure from ground contamination to a MEMBER OF THE PUBLIC at the nearest exclusion area boundary will be within 0.5 rem.

This specification applies to the secondary system liquid waste discharges of radioactive materials from all reactor units to the onsite evaporation ponds. Since the chemical neutralizer tank concentrations will bound concentrations in other secondary waste discharges, surveillance requirements stipulate that sampling and analysis of other secondary waste discharges need be performed only if the sampling and analysis of the contents of the chemical neutralizer tank shows that the neutralizer tank concentration exceeds the specified LLD.

The required detection capabilities for radioactive materials in the secondary liquid waste samples are tabulated in terms of the lower limits of detection (LLDs). Detailed discussion of the LLD, and other detection limits can be found in HASL Procedures Manual, HASL-300 (revised annually), Currie, L. A., "Limits for Qualitative Detection and Quantitative Determination - Application to Radiochemistry," Anal. Chem. 40, 586-93 (1968), and Hartwell, J. K., "Detection Limits for Radioanalytical Counting Techniques," Atlantic Richfield Hanford Company Report ARH-SA-215 (June 1975).

3/4.11.1.2 DOSE

This specification is provided to implement the requirements of Sections II.A, III.A and IV.A of Appendix I, 10 CFR Part 50. The Limiting Condition for Operation implements the guides set forth in Section II.A of Appendix I. The ACTION statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive material in liquid effluents to UNRESTRICTED AREAS will be kept "as low as is reasonably achievable." Also, for fresh water sites with drinking water supplies that can be potentially affected by plant operations, there is reasonable assurance that the operation of the facility will not result in radionuclide concentrations in the finished drinking water that are in excess of the requirements of 40 CFR Part 141. The dose calculation methodology and parameters in the ODCM implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I be shown by calculational procedures based on models and data, such that the actual exposure of a MEMBER OF THE PUBLIC through appropriate pathways is unlikely to be substantially underestimated. The equations specified in the ODCM for calculating the doses due to the actual release rates of radioactive materials in liquid effluents are consistent with

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~~the methodology provided in Regulatory Guide 1.109,~~ "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Revision 1, October 1977 and Regulatory Guide 1.113, "Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I," April 1977.

This specification applies to the release of liquid effluents from each reactor at the site. For units with shared radwaste treatment systems, the liquid effluents from the shared system are proportioned among the units sharing that system.

3/4.11.1.3 LIQUID HOLDUP TANKS

The tanks referred to in this specification include all those outdoor radwaste tanks that are not surrounded by liners, dikes, or walls capable of holding the tank contents and that do not have tank overflows and surrounding area drains connected to the liquid radwaste treatment system.

Restricting the quantity of radioactive material contained in the specified tanks provides assurance that in the event of an uncontrolled release of the tanks' contents, the resulting concentrations would be less than the limits of 10 CFR Part 20, Appendix B, Table II, Column 2, at the nearest potable water supply and the nearest surface water supply in an UNRESTRICTED AREA.

The limit of 60 curies is based on the analyses given in Section 2.4 of the PVNGS FSAR and on the amount of soluble (not gaseous) radioactivity in the ~~Refueling Water Tank in Table 2.4-26.~~

3/4.11.2 GASEOUS EFFLUENTS

~~3/4.11.2.1 DOSE RATE~~ ← See Applicant's SAR

~~This specification is provided to ensure that the dose at any time at and~~ beyond the SITE BOUNDARY from gaseous effluents from all units on the site will be within the annual dose limits of 10 CFR Part 20 to UNRESTRICTED AREAS. The annual dose limits are the doses associated with the concentrations of 10 CFR Part 20, Appendix B, Table II, Column 1. These limits provide reasonable assurance that radioactive material discharged in gaseous effluents will not result in the exposure of a MEMBER OF THE PUBLIC in an UNRESTRICTED AREA, either within or outside the SITE BOUNDARY, to annual average concentrations exceeding the limits specified in Appendix B, Table II of 10 CFR Part 20 (10 CFR Part 20.106(b)). For MEMBERS OF THE PUBLIC who may at times be within the SITE BOUNDARY, the occupancy of that MEMBER OF THE PUBLIC will usually be sufficiently low to compensate for any increase in the atmospheric diffusion factor above that for the SITE BOUNDARY. Examples of calculations for such MEMBERS OF THE PUBLIC, with the appropriate occupancy factors, shall be given in the ODCM. The specified release rate limits restrict, at all times, the corresponding gamma and beta dose rates above background to a MEMBER OF THE ~~PUBLIC at or beyond the SITE BOUNDARY to less than or equal to 500 mrems/year~~

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~~GASEOUS RADWASTE TREATMENT~~ (Continued)

This specification applies to the release of radioactive materials in gaseous effluents from each reactor unit at the site.

The minimum analysis frequency of 4/M (i.e. at least 4 times per month at intervals no greater than 9 days and a minimum of 48 times a year) is used for certain radioactive gaseous waste sampling in Table 4.11-2. This will eliminate taking double samples when quarterly and weekly samples are required at the same time.

3/4.11.2.5 EXPLOSIVE GAS MIXTURE

This specification is provided to ensure that the concentration of potentially explosive gas mixtures contained in the waste gas holdup system is maintained below the flammability limits of hydrogen and oxygen. (Automatic control features are included in the system to prevent the hydrogen and oxygen concentrations from reaching these flammability limits. These automatic control features include isolation of the source of hydrogen and/or oxygen, automatic diversion to recombiners, or injection of dilutants to reduce the concentration below the flammability limits.) Maintaining the concentration of hydrogen and oxygen below their flammability limits provides assurance that the releases of radioactive materials will be controlled in conformance with the requirements of General Design Criterion 60 of Appendix A to 10 CFR Part 50.

3/4.11.2.6 GAS STORAGE TANKS

This specification considers postulated radioactive releases due to a waste gas system leak or failure, and limits the quantity of radioactivity contained in each pressurized gas storage tank in the GASEOUS RADWASTE SYSTEM to assure that a release would be substantially below the guidelines of 10 CFR Part 100 for a postulated event.

Restricting the quantity of radioactivity contained in each gas storage tank provides assurance that in the event of an uncontrolled release of the tank's contents, the resulting total body exposure to a MEMBER OF THE PUBLIC at the nearest exclusion area boundary will not exceed 0.5 rem. This is consistent with Standard Review Plan 11.3, Branch Technical Position ETSB 11-5, "Postulated Radioactive Releases Due to a Waste Gas System Leak or Failure," ~~in NUREG-0800, July 1981~~.

3/4.11.3 SOLID RADIOACTIVE WASTE

R See Applicant's SAR

~~This specification addresses the requirements of General Design Criterion 60 of Appendix A to 10 CFR Part 50. The process parameters included in establishing the PROCESS CONTROL PROGRAM may include, but are not limited to waste type, waste pH, waste/liquid/solidification agent/catalyst ratios, waste oil content, waste principal chemical constituents, and mixing and curing times.~~

3/4.11.4 TOTAL DOSE

R See Applicant's SAR.

~~This specification is provided to meet the dose limitations of 40 CFR Part 190 that have been incorporated into 10 CFR Part 20 by 46 FR 18525. The specification requires the preparation and submittal of a Special Report whenever the calculated doses from plant generated radioactive effluents and~~

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BASES

3/4.12.1 MONITORING PROGRAM

See Applicant's SAR.

~~The radiological environmental monitoring program required by this~~ specification provides representative measurements of radiation and of radioactive materials in those exposure pathways and for those radionuclides that lead to the highest potential radiation exposures of MEMBERS OF THE PUBLIC resulting from the station operation. This monitoring program implements Section IV.B.2 of Appendix I to 10 CFR Part 50 and thereby supplements the radiological effluent monitoring program by verifying that the measurable concentrations of radioactive materials and levels of radiation are not higher than expected on the basis of the effluent measurements and the modeling of the environmental exposure pathways. Guidance for this monitoring program is provided by the Radiological Assessment Branch Technical Position on Environmental Monitoring. The initially specified monitoring program will be effective for at least the first 3 years of commercial operation. Following this period, program changes may be initiated based on operational experience.

The required detection capabilities for environmental sample analyses are tabulated in terms of the lower limits of detection (LLDs). The LLDs required by Table 4.12-1 are considered optimum for routine environmental measurements in industrial laboratories. It should be recognized that the LLD is defined as an *a priori* (before the fact) limit representing the capability of a measurement system and not as an *a posteriori* (after the fact) limit for a particular measurement.

Detailed discussion of the LLD, and other detection limits, can be found in HASL Procedures Manual, HASL-300 (revised annually), Currie, L. A., "Limits for Qualitative Detection and Quantitative Determination - Application to Radiochemistry," Anal. Chem. 40, 586-93 (1968), and Hartwell, J. K., "Detection Limits for Radioanalytical Counting Techniques," Atlantic Richfield Hanford Company Report ARH-SA-215 (June 1975).

BASES

3/4.12.2 LAND USE CENSUS

R See Applicant's SAR.

~~This specification is provided to ensure that changes in the use of areas at and beyond the SITE BOUNDARY are identified and that modifications to the radiological environmental monitoring program are made if required by the results of this census. The best information from the door-to-door survey, from aerial survey or from consulting with local agricultural authorities shall be used. This census satisfies the requirements of Section IV.B.3 of Appendix I to 10 CFR Part 50. Restricting the census to gardens of greater than 50 m^2 provides assurance that significant exposure pathways via leafy vegetables will be identified and monitored since a garden of this size is the minimum required to produce the quantity (26 kg/year) of leafy vegetables assumed in Regulatory Guide 1.109 for consumption by a child. To determine this minimum garden size, the following assumptions were made: (1) 20% of the garden was used for growing broad leaf vegetation (i.e., similar to lettuce and cabbage), and (2) a vegetation yield of 2 kg/m^2.~~

3/4.12.3 INTERLABORATORY COMPARISON PROGRAM

R See Applicant's SAR

~~The requirement for participation in an approved Interlaboratory Comparison Program is provided to ensure that independent checks on the precision and accuracy of the measurements of radioactive material in environmental sample matrices are performed as part of the quality assurance program for environmental monitoring in order to demonstrate that the results are valid for the purposes of Section IV.B.2 of Appendix I to 10 CFR Part 50.~~

SECTION 5.0

DESIGN FEATURES

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5.0 DESIGN FEATURES

5.1 SITE

SITE AND EXCLUSION BOUNDARIES

5.1.1 The site and exclusion boundaries shall be as shown in Figure 5.1-1.

LOW POPULATION ZONE

5.1.2 The low population zone shall be as shown in Figure 5.1-2.

GASEOUS RELEASE POINTS

5.1.3 The gaseous release points shall be as shown in Figure 5.1-3.

5.2 CONTAINMENT

CONFIGURATION

See Applicant's SAR.

5.2.1 ~~The reactor containment building is a steel lined, prestressed concrete building of cylindrical shape, with a dome roof and having the following design features:~~

~~a. Nominal inside diameter = 146 feet.~~

~~b. Nominal inside height = 206.5 feet.~~

~~c. Minimum thickness of concrete walls = 3 feet, 8 inches.~~

~~d. Minimum thickness of concrete roof = 3 feet, 8 inches.~~

~~e. Minimum thickness of concrete floor pad = 10.5 feet.~~

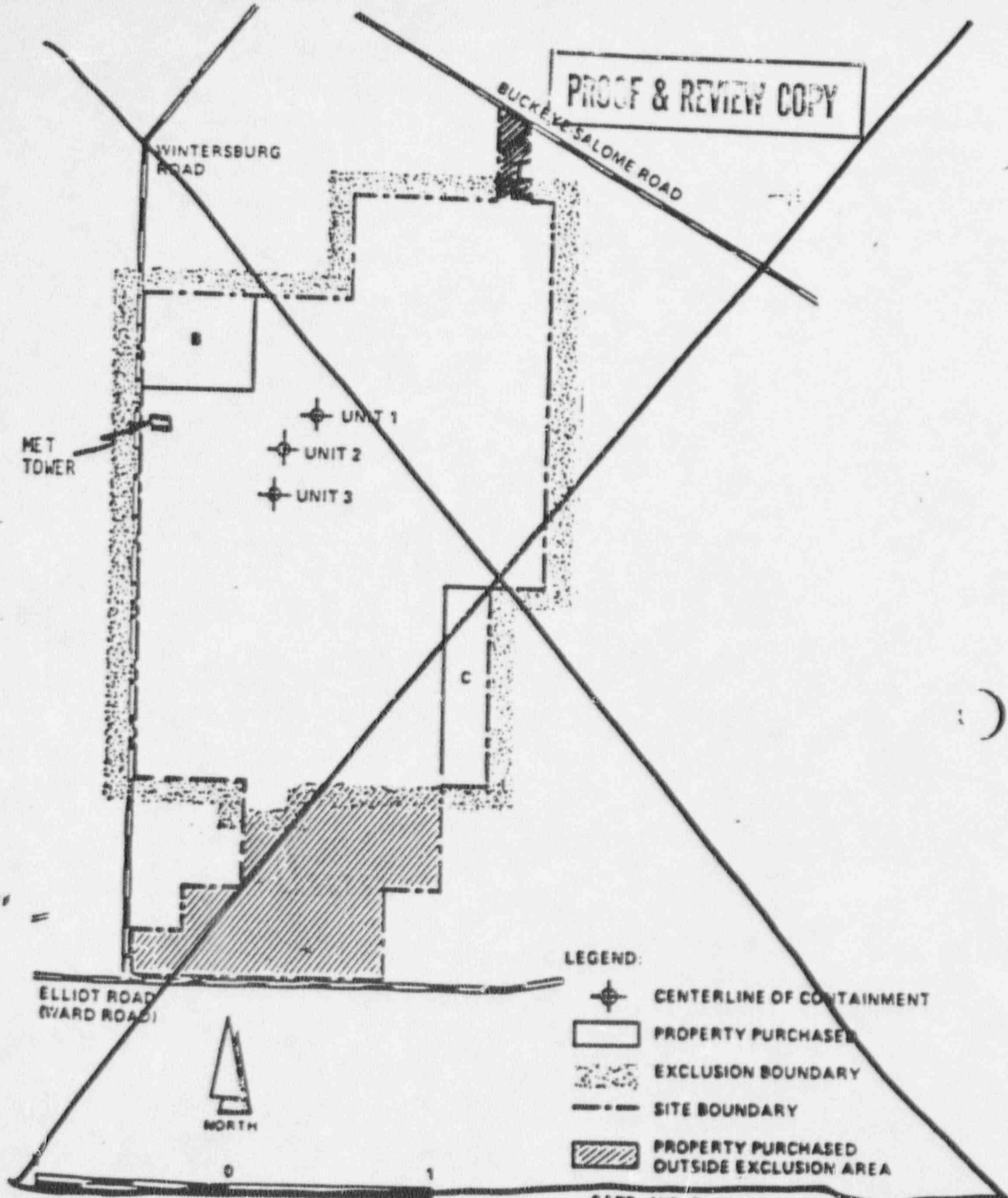
~~f. Nominal thickness of steel liner = 0.25 inch.~~

~~g. Net free volume = 2.6×10^6 cubic feet.~~

DESIGN PRESSURE AND TEMPERATURE

See Applicant's SAR.

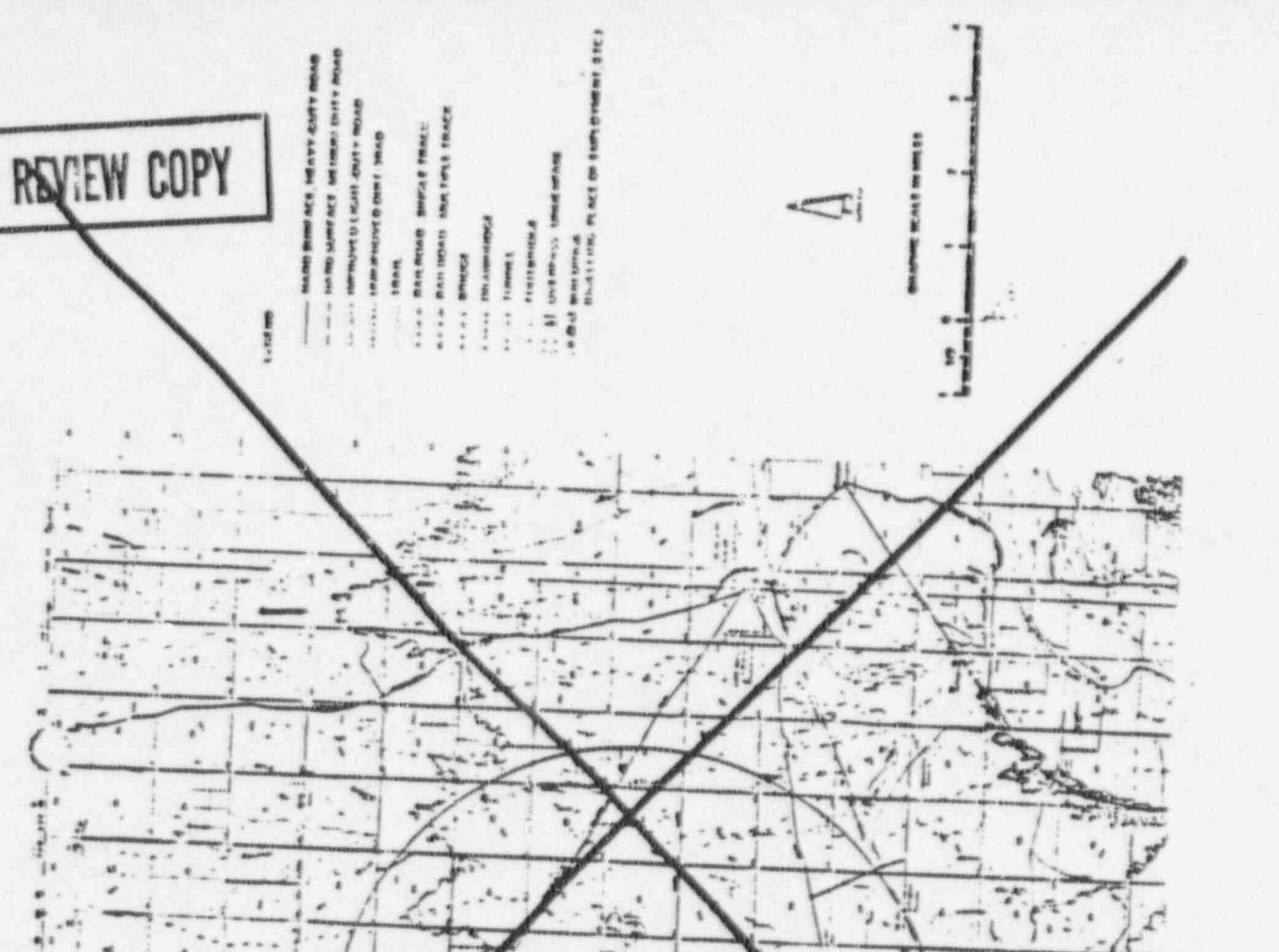
5.2.2 ~~The reactor containment building is designed and shall be maintained for a maximum internal pressure of 60 psig and a temperature of 300°F.~~



SITE AND EXCLUSION BOUNDARIES

FIGURE 5.1-1

CESSAR80-NSSS-STS ~~PALO VERDE UNIT 1~~ SEE APPLICANTS SAR



SEE APPLICANTS SAR

LOW POPULATION ZONE
~~PALO VERDE NUCLEAR GENERATING STATION~~

FIGURE 5.1-2

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5.3 REACTOR CORE

FUEL ASSEMBLIES

a maximum of

1900

5.3.1 The reactor core shall contain 241 fuel assemblies with each fuel assembly containing 236 fuel rods or burnable poison rods clad with Zircaloy-4. Each fuel rod shall have a nominal active fuel length of 150 inches and contain a maximum total weight of approximately ~~[illegible]~~ grams uranium. Each burnable poison rod shall have a nominal active poison length of 136 inches. The initial core loading shall have a maximum enrichment of ~~[illegible]~~ 3.30 weight percent U-235. Reload fuel shall be similar in physical design to the initial core loading and shall have a maximum enrichment of 4 weight percent U-235.

CONTROL ELEMENT ASSEMBLIES

5.3.2 The reactor core shall contain 76 full-length and 13 part-length control element assemblies.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The Reactor Coolant System is designed and shall be maintained:

a. In accordance with the code requirements specified in Section 5.2 of the FSAR with allowance for normal degradation pursuant of the applicable surveillance requirements,

b. For a pressure of 2500 psia, and

c. For a temperature of 650°F, except for the pressurizer which is 700°F.

VOLUME

5.4.2 The total water and steam volume of the Reactor Coolant System is 13,900 + 300/-0 cubic feet at a nominal T_{avg} of 593°F.

5.5 METEOROLOGICAL TOWER LOCATION

5.5.1 The meteorological tower shall be located as shown on Figure 5.1-1.

5.6 FUEL STORAGE

~~5.6.1~~ CRITICALITY See Applicant's SAR.

5.6.1 ~~The spent fuel storage racks are designed and shall be maintained with:~~

a. ~~A k_{eff} equivalent to less than or equal to 0.95 when flooded with~~ unborated water, which includes a conservative allowance of ~~2.6%~~ delta k/k for uncertainties as described in Section 9.1 of the FSAR.

b. A nominal 9.5 inch center-to-center distance between fuel assemblies placed in the storage racks in a high density configuration.

5.6.1.2 The k_{eff} for new fuel for the first core loading stored dry in the spent fuel storage racks shall not exceed 0.98 when aqueous foam moderation ~~is assumed.~~

DRAINAGE See Applicant's SAR.

5.6.2 ~~The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 137 feet - 6 inches.~~

CAPACITY See Applicant's SAR.

5.6.3 ~~The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 1329 fuel assemblies.~~

5.7 COMPONENT CYCLIC OR TRANSIENT LIMITS

5.7.1 The components identified in Table 5.7-1 are designed and shall be maintained within the cyclic or transient limits of Tables 5.7-1 and 5.7-2.

TABLE 5.7-1

COMPONENT CYCLIC OR TRANSIENT LIMITS

COMPONENT	CYCLIC OR TRANSIENT LIMIT	DESIGN CYCLE OR TRANSIENT
Reactor Coolant System	500 system heatup and cooldown cycles at rates ≤ 100°F/hr.	Heatup cycle - Temperature from ≤ 70°F to ≥ 565°F; cooldown cycle - Temperature from ≥ 565°F to ≤ 70°F.
	500 pressurizer heatup and cooldown cycles at rates ≤ 200°F/hr.	Heatup cycle - Pressurizer temperature from ≤ 70°F to ≥ 653°F; cooldown cycle - Pressurizer temperature from ≥ 653°F to ≤ 70°F.
	10 hydrostatic testing cycles.	RCS pressurized to 3125 psia with RCS temperature between 120°F and 400°F.
	480 reactor trip cycles, turbine trip cycles, and loss of reactor coolant flow.	Includes combinations of reactor trips due to operator errors, equipment malfunctions, and total loss of reactor coolant flow.
	200 seismic stress cycles.	Subjection to a seismic event equal to one-half the design basis earthquake (DBE).
	1 complete loss of secondary pressure cycle.	Loss of secondary pressure from either steam generator due to a complete double-ended break of a steam generator steam or feedwate nozzle.

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TABLE 5.7-1 (Continued)

COMPONENT CYCLIC OR TRANSIENT LIMITS

COMPONENT	CYCLIC OR TRANSIENT LIMIT	DESIGN CYCLE OR TRANSIENT
	200 primary system leak test cycles	Leak test primary system at a pressure of 2250 psia at a temperature from 120°F to 400°F.
Pressurizer Spray Nozzle	Calculate usage factor per Table 5.7-2.	Main spray (less than four RCP operating) with fluid $\Delta T_m > 200°F$. Auxiliary spray with fluid $\Delta T_a > 200°F$.

ΔT_m = The difference in temperature between the pressurizer and main spray water as adjusted by the instrument correction factor.

ΔT_a = The difference in temperature between the pressurizer and Auxiliary spray water as adjusted by the instrument correction factor.

TABLE 5.7-2

PRESSURIZER SPRAY NOZZLE USAGE FACTOR

SEE APPLICANT'S SAR

Main Spray				Auxiliary Spray			
ΔT_m	N_A	N	N/N_A	ΔT_a	N_A	N	N/N_A
201-250	7900			201-250	50000		
251-300	4500			251-300	2200		
301-350	2900			301-350	1300		
351-400	1900			351-400	850		
401-450	1200			401-450	550		
451-500	850			451-500	375		
501-550	555			501-550	225		
				551-600	150		

$\Sigma N/N_A$ = ______ $\Sigma N/N_A$ = ______

Cumulative Usage Factor

$\Sigma N/N_A$ (Main Spray) ______

$\Sigma N/N_A$ (Aux. Spray) ______

Total ______ = Cumulative Usage Factor

PALO VERDE ONLY

CESSAR 80-NSSS-STS

5-58

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TABLE 5.7-2 (Continued)

Where:

$\Delta T_a = (T_{101} - T_{229}) + 60$

$\Delta T_m = (T_{101} - T_{103^* \text{ or } 104^*}) + 70$

NA = Allowable number of spray cycles

N = Number of cycles in ΔT range indicated

Calculational Method:

1. The spray cycle is defined as any initiation and termination of main or auxiliary spray flow throughout the pressurizer spray nozzle.

2. If the difference between pressurizer water temperature and the spray water temperature exceeds 200°F each spray cycle and the corresponding temperature difference is logged.

3. The spray nozzle usage factor shall be calculated as follows:

   A. Fill in Column "N" above.

   B. Calculate "N/N_A" (Divide N by N_A).

   C. Add Column "N/N_A" to find $\Sigma N/N_A$.

   $\Sigma N/N_A$ is the cumulative spray nozzle usage factor. If the cumulative usage factor is equal to or less than 0.65 no further action is required.

4. If the cumulative usage factor exceeds 0.65, subsequent pressurizer spray operation shall continue to be monitored and an engineering evaluation of nozzle fatigue shall be performed within 90 days. The evaluation shall determine that the nozzle remains acceptable for additional service beyond the 90 day period or subsequent spray operation shall be restricted so that the difference between the pressurizer water temperature and the spray water temperature shall be limited to less than or equal to 200°F when spray is operated.

*Use lower of two temperatures.

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SECTION 6.0

ADMINISTRATIVE CONTROLS

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~~6.1 RESPONSIBILITY~~

6.1.1 The PVNGS Plant Manager shall be responsible for overall unit operation and shall delegate in writing the succession to this responsibility during his absence.

6.1.2 The Shift Supervisor, or during his absence from the Control Room, a designated SRO, shall be responsible for the Control Room command function. A management directive to this effect, signed by the Vice President-Nuclear Production shall be reissued to all station personnel on an annual basis.

6.2 ORGANIZATION

OFFSITE

6.2.1 The offsite organization for unit management and technical support shall be as shown in Figure 6.2-1.

UNIT STAFF

6.2.2.1 The unit organization shall be as shown in Figure 6.2-2 and:

a. Each on-duty shift shall be composed of at least the minimum shift crew composition shown in Table 6.2-1.

b. At least one licensed Reactor Operator shall be in the Control Room when fuel is in the reactor. In addition, while the reactor is in MODE 1, 2, 3, or 4, at least one licensed Senior Reactor Operator shall be in the Control Room.

c. A radiation protection technician* shall be onsite when fuel is in the reactor.

d. All CORE ALTERATIONS shall be observed and directly supervised by either a licensed Senior Reactor Operator or Senior Reactor Operator Limited to Fuel Handling who has no other concurrent responsibilities during this operation.

e. A site Fire Team of at least five members shall be maintained onsite at all times*. The Fire Team shall not include the Shift Supervisor, the STA, nor the 3 other members of the minimum shift crew necessary for safe shutdown of the unit and any personnel required for other essential functions during a fire emergency.

6.2.2.2 The unit staff working hours shall be as follows:

a. Administrative procedures shall be developed and implemented to limit the working hours of unit staff who perform safety-related functions; e.g., Senior Reactor Operators, Reactor Operators, radiation protection technicians, auxiliary operators, and key maintenance personnel.

INSERT

*The radiation protection technician and Fire Team composition may be less than the minimum requirements for a period of time not to exceed 2 hours, in order to accommodate unexpected absence, provided immediate action is taken ~~to fill the required positions.~~

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INSERT 2 pages

[redacted]

6.1 RESPONSIBILITY

See Applicant's SAR.

6.2 ORGANIZATION (includes Figures 6.2-1 and 6.2-2)

See Applicant's SAR.

6.3 UNIT STAFF QUALIFICATION

See Applicant's SAR.

6.4 TRAINING

See Applicant's SAR.

6.5 REVIEW AND AUDIT

See Applicant's SAR.

Ⓒ 6.6 REPORTABLE EVENT ~~OCCURRENCE~~ ACTION

See Applicant's SAR.

6.7 SAFETY LIMIT VIOLATION

See Applicant's SAR.

6.8 PROCEDURES AND PROGRAMS

See Applicant's SAR.

6.9 REPORTING REQUIREMENTS

See Applicant's SAR.

6.10 RECORD RETENTION

See Applicant's SAR.

6.11 RADIATION PROTECTION PROGRAM

See Applicant's SAR.

6.12 HIGH RADIATION AREA

See Applicant's SAR.

[redacted]

6.13 PROCESS CONTROL PROGRAM

See Applicant's SAR.

6.14 OFFSITE DOSE CALCULATION MANUAL

See Applicant's SAR.

6.15 MAJOR CHANGES TO RADIOACTIVE LIQUID, GASEOUS AND SOLID WASTE TREATMENT SYSTEMS

See Applicant's SAR.

6.16 PRE-PLANNED ALTERNATE SAMPLING PROGRAM

See Applicant's SAR.