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Appendix 3 Criticality

3.1 Introduction

The staff criticality assessment includes both a more classical deterministic study and a qualitative risk study. The conclusion in Chapter 3 of this report that criticality is not a risk significant event is based upon consideration of both of these studies. The deterministic study was used to define the possible precursor scenarios and any mitigative actions. The risk study considered whether the identified scenarios are credible and whether any of the identified compensatory measures are justified given the probability of the initiating scenario. This appendix combines both the risk study, the consequences, and the report on the deterministic criticality assessment into one location for easy reference.

3.2 Qualitative Risk Study

3.2.1 Criticality in Spent Fuel Pool

Due to the processes involved and lack of data, it was not possible to perform a quantitative risk assessment for criticality in the spent fuel pool. Enclosed as section 3.2.2 is a deterministic study in which the staff performs an evaluation of the potential scenarios that could lead to criticality and identified those that are credible. In this section the staff provides its qualitative assessment of risk due to criticality in the SFP, and its conclusions that the potential risk from SFP criticality is sufficiently small.

In the report enclosed in section 3.2.2, the staff assessed the various potential scenarios that could result in inadvertent criticality. This assessment identified two scenarios as credible, which are listed below.

(1) A compression or buckling of the stored assemblies could result in a more optimum geometry (closer spacing) and thus create the potential for criticality (see the NRC staff report "Assessment of the Potential for Criticality in Decommissioned Spent Fuel Pools," at the end of Appendix 3). Compression is not a problem for high-density PWR or BWR racks because they have sufficient fixed neutron absorber plates to mitigate any reactivity increase, nor is it a problem for low-density PWR racks if soluble boron is credited. But compression of a low-density BWR rack could lead to a criticality since BWR racks contain no soluble or solid neutron absorbing material. High-density racks are those that rely on both fixed neutron absorbers and geometry to control reactivity. Low-density racks rely solely upon geometry for reactivity control. In addition, all PWR pools are borated, whereas BWR pools contain no soluble absorbing material. If both PWR and BWR pools were borated, criticality would not be achievable for a compression event.

(2) If the stored assemblies are separated by neutron absorber plates (e.g., Boral or Boraflex), loss of these plates could result in a potential for criticality for BWR pools. For PWR pools, the soluble boron would be sufficient to maintain subcriticality. The absorber plates are generally enclosed by cover plates (stainless steel or aluminum alloy). The tolerances within a cover plate tend to prevent any appreciable

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fragmentation and movement of the enclosed absorber material. The total loss of the welded cover plate is not considered feasible.

Boraflex has been found to degrade in spent fuel pools due to gamma radiation and exposure to the wet pool environment. For this reason, the NRC issued Generic Letter 96-04 to all holders of operating licenses, on Boraflex degradation in spent fuel storage racks. Each addressee that uses Boraflex was requested to assess the capability of the Boraflex to maintain a 5% subcriticality margin and to submit to the NRC proposed actions to monitor the margin or confirm that this 5% margin can be maintained for the lifetime of the storage racks. Many licensees subsequently replaced the Boraflex racks in their pools or reanalyzed the criticality aspects of their pools, assuming no reactivity credit for Boraflex.

Other potential criticality events, such as loose debris of pellets or the impact of water or firefighting foam (adding neutron moderation) during personnel actions in response to accidents was discounted due to the basic physics and neutronic properties of the racks and fuel, which would preclude criticality conditions being reached with any creditable likelihood. For example, without moderation, fuel at current enrichment limits (no greater than 5 wt% U-235) cannot achieve criticality, no matter what the configuration. If it is assumed that the pool water is lost, a reflooding of the storage racks with unborated water or fire-fighting foam may occur due to personnel actions. However, both PWR and BWR storage racks are designed to remain subcritical if moderated by unborated water in the normal configuration. The phenomenon of a peak in reactivity due to low-density (optimum) moderation (fire-fighting foam) is not of concern in spent fuel pools since the presence of relatively weak absorber materials such as stainless steel plates or angle brackets is sufficient to preclude neutronic coupling between assemblies. Therefore, personnel actions to refill a drained spent fuel pool containing undeformed fuel assemblies would not create the potential for a criticality. Thus, the only potential scenarios described above in 1 and 2 involve crushing of fuel assemblies in low density racks or degradation of Boraflex over long periods in time.

To gain qualitative insights on the criticality events that are credible, the staff considered the sequences of events that must occur. For scenario 1, above this would require a heavy load drop into the a low density racked BWR pool compressing assemblies. From appendix 2 on heavy load drop, the likelihood of a heavy load drop from a single failure proof crane is approximately 2E-6 per year, assuming 100 cask movements per year at the decommissioning facility. From the load path analysis done for that appendix it was estimated that the load could be over or near the pool between 25% and 5% of the movement path length, dependent on plant specific layout specifics. The additional frequency reduction in the appendix to account for the fraction of time that the heavy load is lifted high enough to damage the pool liner is not applicable here because the fuel assemblies could be crushed without the same impact velocity being required as for the pool liner. Therefore, if we assume 10% load path vulnerability, we observe a potential initiating frequency for crushing of approximately 2E-7 per year (based upon 100 lifts per year). Criticality calculations show that even if the low density BWR assemblies were crushed by a transfer cask, it is "highly unlikely" that a configuration would be reached that would result in a severe reactivity event, such as a steam explosion which could damage and drain the spent fuel pool. The staff judges the chances of such a criticality event to be well below 1 chance in 100 even given that the transfer cask drops directly onto the assemblies. This would put the significant criticality likelihood well below 1E-8 per year, which justifies its exclusion from further consideration.

Deformation of the low density BWR racks by the dropped transfer cask was shown to most likely not result in any criticality events. However, if some mode of criticality was to be induced by the dropped transfer cask it would more likely be a small return to power for a very localized region, rather than the severe response discussed the above paragraph. This minor type of event would have essentially no offsite (or onsite) consequences since the reaction's heat would be removed by localized boiling in the pool and water would provide shielding to the site operating staff. The reaction could be terminated with relative ease by the addition of boron to the pool. Therefore, the staff believes that qualitative (as well as some quantitative) assessment of scenario 1 demonstrates that it poses no significant risk to the public from SFP operation during the period that the fuel remains stored in the pool.

With respect to scenario #2 from above, (the gradual degradation of the Boraflex absorber material in high density storage racks), there is currently not sufficient data to quantify the likelihood of criticality occurring due to its loss. However the current programs in place at operating plants to assess the condition of the Boraflex, and take remedial action if necessary provide sufficient confidence that pool reactivity requirements will be satisfied . In order to meet the RG 1.174 safety principle of maintaining sufficient safety margins, the staff judges that continuation of such programs into the decommissioning phase will required at all plants until all high density racks are removed from the SFP.

Additionally, to provide an element of defense in depth, the staff believes that inventories of boric acid be maintained on site, to respond to scenarios where loss of pool inventories have to be responded to by makeup of unborated water at PWR sites. The staff will also require that procedures be available to provide guidance to the operating staff as to when such boron addition may be beneficial.

Based upon the above conclusions and staff requirements, we believe that qualitative risk insights demonstrate conclusively that SFP criticality poses so meaningful risk to the public.

3.2.2 Deterministic Criticality Study

This section includes a copy of the report entitled "Assessment of the Potential for Criticality in Decommissioned Spent Fuel Pools" which is a deterministic study of the potential for spent fuel pool criticality.

Assessment of the Potential for Criticality in Decommissioned Spent Fuel Pools

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Introduction

The staff has performed a series of calculations to assess the potential for a criticality accident in the spent fuel pool of a decommissioned nuclear power plant. This work was undertaken to support the staff's efforts to develop a decommissioning rule. Unlike operating spent fuel storage pools, decommissioned pools will have to store some number of spent fuel assemblies which have not achieved full burnup potential for extended periods of time which were used in the final operating cycle of the reactor. Operating reactors typically only store highly reactive assemblies for short periods of time. These assemblies constitute approximately one third of the assemblies in the final operating cycle of the reactor. These assemblies are more reactive than those assemblies normally stored in the pool which have undergone full burnup. Operating reactors typically only store similarly reactive assemblies for short periods of time during refueling or maintenance outages. As we will see in this report, the loss of geometry alone could cause a criticality accident unless some mitigative measures are in place.

When spent fuel pools were originally conceived, they were intended to provide short term storage for a relatively small number of assemblies while they decayed for a period of time sufficient to allow their transport to a long term storage facility. Because a long term storage facility is not available, many reactor owners have had to change the configuration of their spent fuel pools on one or, in some cases, several occasions. This practice has led to a situation where there are many different storage configurations at U.S. plants utilizing some combination of geometry, burnup, fixed poisons, and boration, to safely store spent fuel.

The current state of spent fuel pools significantly complicates the task of generically analyzing potential spent fuel pool storage configurations. Therefore, the staff decided to take a more phenomenalogical approach to the analysis. Rather than trying to develop specific scenarios for the different types of loading configurations, we decided to analyze storage rack deformation and degradation by performing bounding analyses using typical storage racks. The results of these analyses will be used to formulate a set of generic conclusions regarding the physical controls necessary to prevent criticality. The impact of five pool storage assumptions on the conclusions in this report will be discussed throughout the text. Furthermore, for the purposes of this work, it is assumed that the postulated criticality event is unrecoverable when the water level reaches the top of the fuel. This means that events such as a loss of water leading to a low density optimal moderation condition caused by firefighting equipment will not be considered.

It is important to reinforce the point that these analyses are intended as a guide only and will be used to evaluate those controls that are either currently in place or will need to be added to maintain subcriticality. These analyses will not be used to develop specific numerical limits which must be in place to control criticality as they cannot consider all of the possible plant specific variables. We will, however, define the controls that would be effective either individually or in combination to preclude a criticality accident.

Description Of Methods

The criticality analyses were performed with three-dimensional Monte Carlo methods using ENDF/B-V based problem specific cross sections (Ref. 1). Isotopic inventories were predicted using both one- and two-dimensional transport theory based methods with point depletion. SCALE 4.3 (Ref. 2) was used to perform the Monte Carlo, one-dimensional transport, cross section processing, and depletion calculations. Specifically, the staff used KENO-VI, NITAWL-1, BONAMI, XSDRN, and ORIGEN. The two-dimensional transport theory code NEWT (Ref. 3) was used for Boiling Water Reactor (BWR) lattice depletion studies. NEWT uses the method of characteristics to exactly represent the two-dimensional geometry of the problem. NEWT uses ORIGEN for depletion. Cross section data were tracked and used on a pin cell basis for the BWR assessments. The staff developed post processing codes to extract the information from NEWT and create an input file suitable for use with SCALE. Both the 238 and the 44 group ENDF/B-V based libraries were used in the project. Refer to Sample Input Deck at the end of Appendix 7 for a listing of one of the input decks used in this analysis. SCALE has been extensively validated for these types of assessments. (see References 4, 5, and 6)

Problem Definition

Compression (or expansion) events were analyzed in two ways. First, the assembly was assumed to crush equally in the x and y directions (horizontal plane). Analyses were performed with and without the fixed absorber panels *without* soluble boron and with fuel at the most reactive point allowed for the configuration. In these cases, the fuel pin pitch was altered to change the fuel to moderator ratio. These scenarios are intended to simulate the crushing (or expansion) of a high density configuration when little or no rack deformation is necessary to apply force to the fuel assembly. The scenarios are also applicable to low density rack deformation in which the rack structure collapses to the point at which force is applied to the assemblies. The second type of compression event involved changing the intra-assembly spacing, but leaving the basic lattice geometry unchanged. These simulations were intended to simulate compression events in which the force applied to the rack is insufficient to compress the assembly.

Discussion Of Results

Several observations are common to both Pressurized Water Reactor (PWR) and BWR rack designs. First of all, poisoned racks should remain subcritical during all compression type events assuming that the poison sheeting remains in place (in other words, that it compresses with the rack and does not have some sort of brittle failure). Secondly, criticality cannot be precluded by design following a compression event for low density, unpoisoned (referring to both soluble and fixed poisons) storage racks.

PWR Spent Fuel Storage Racks

The analyses and this discussion will differentiate between high and low density storage. High density storage is defined as racks that rely on both fixed poison sheets and geometry to control reactivity and low density storage relies solely upon geometry for reactivity control. The results of the analyses for the high density storage racks is summarized in Figure 1. When discussing Figure 1 it should be noted that the analyses supporting Figure 1 were performed without soluble boron and with fuel at the most reactive point allowed for the rack. These assumptions represent a significant conservatism of at least 20 percent delta-k. Figure 1 demonstrates that even with compression to an optimal geometric configuration, criticality is prevented by design (for these scenarios we are not trying to maintain a k_{eff} less than 0.95). The poison sheeting, boral in this case, is sufficient to keep the configuration subcritical.

The results for the low density storage rack are given in Figure 2. As can be seen, criticality cannot be entirely ruled out on the basis of geometry alone. Therefore, we examined the conservatism implicit in the methodology and assessed whether there is enough margin to not require any additional measures for criticality control. There are two main sources of conservatism in the analyses; using fuel at the most reactive state allowed for the configuration and not crediting soluble boron. By relaxing the assumption that all of the fuel is at its peak expected reactivity, we have demonstrated by analyzing several sample storage configurations that the rack eigenvalue can be reduced to approximately 0.998 (see Table 1). The storage configurations analyzed included placing a most reactive bundle every second, fourth, sixth and eighth storage cell (see Figure 3). The assemblies used between the most reactive assembly were defined by burning the 5 w/o U_{235} enriched Westinghouse 15x15 assembly to 55 GWD/MTU which is a typical discharge burnup for an assembly of this type. This study did not examine all possible configurations so this value should be taken as an estimate only. However, the study does suggest that scattering the most reactive fuel throughout the pool would substantially reduce the risk of a criticality accident. It is difficult to entirely relax the assumption of no soluble boron in the pool, but its presence will allow time for recovery actions during an event that breaches the SFP liner and compresses the rack but does not rapidly drain the pool.

Although not all-inclusive because all fuel and rack types were not explicitly considered, the physical controls that were identified are generically applicable. The fuel used in this study is a Westinghouse 15x15 assembly enriched to 5 w/o U_{235} with no burnable absorbers. The Westinghouse 15x15 assembly has been shown by others (Ref. 7) to be the most reactive PWR fuel type when compared to a large number of different types of PWR fuel. Furthermore, the use of 5 w/o U_{235} enriched fuel will bound all available fuel types because it represents the maximum allowed enrichment for commercial nuclear fuel.

BWR Spent Fuel Storage Racks

In these analyses, we differentiated between high and low density BWR racks. The conservatism inherent in the analyses must be considered (for BWR racks, the use of the most reactive fuel allowed only) when considering the discussion of these results. The results of the analyses of high density BWR racks are given in Figure 4. As can be seen, criticality is prevented by design for the high density configurations. The poison sheets remain reasonably intact following the postulated compression event. The poison sheeting (in this case Boraflex) is sufficient to maintain subcriticality.

The results of the low density BWR rack analyses are shown in Figure 5. Here, as with the PWR low density racks, criticality cannot be prevented by design. Once again we assessed the impact of eliminating some of the conservatism in the analyses which in the case of BWR storage is only related to the reactivity of the assembly. Analyses were performed placing a most reactive assembly in every second, fourth, sixth and eighth storage cell. The assemblies placed between the most reactive assemblies were defined by burning the 4.12 w/o enriched

General Electric (GE) 12 assembly to 50 GWd/MTU. These analyses demonstrate that it is possible to reduce the rack eigenvalue to approximately 1.009 (see Table 1). As previously mentioned, this study did not include all possible configurations so this value should be taken as an estimate only. Because BWR pools are not borated, there is no conservatism from the assumption of no soluble boron.

Boraflex degradation is another problem that is somewhat unique to BWR spent fuel storage racks. This is true because of the fact that BWR storage pools do not contain soluble boron that provides the negative reactivity in PWR pools to offset the positive effect of Boraflex degradation. Therefore, some compensatory measures need to be in place to provide adequate assurance that Boraflex degradation will not contribute to a criticality event. In operating reactor spent fuel pools that use Boraflex, licensees use some sort of surveillance program to ensure that the 5 percent subcritical margin is maintained. These programs should be continued during and following decommissioning. No criticality calculations were performed for this study to assess Boraflex degradation because it is conservatively assumed that the loss of a substantial amount of Boraflex will most likely lead to a criticality accident.

These analyses are not all inclusive, but we believe that the physical controls identified are generically applicable. We examined all of the available GE designed BWR assemblies for which information was available and identified the assembly used in the study to have the largest K_{inf} in the standard cold core geometry (in other words, in the core with no control rods inserted at ambient temperature) at the time of peak reactivity. This assembly was a GE12 design (10x10 lattice) enriched to an average value of 4.12 w/o U_{235}. Only the dominant part of the lattice was analyzed and it was assumed to span the entire length of the assembly. This conservatism plus the fact that the assembly itself is highly enriched and designed for high burnup operation has led the staff to conclude that these analyses are generically applicable to BWR spent fuel storage pools.

Conclusions

One scenario that has been identified which could lead to a criticality event is a heavy load drop or some other event that compresses a low density rack filled with spent fuel at its peak expected reactivity. This event is somewhat unique to decommissioned reactors because there are more low burnup (high reactivity) assemblies stored in the spent fuel pool that were removed from the core following its last cycle of operation, than in a SFP at an operating plant.

To address the consequences of the compression of a low density rack, there are two strategies that could be used, either individually or in combination. First, the most reactive assemblies (most likely the fuel from the final cycle of operation) could be scattered throughout the pool, or placed in high density storage if available. Second, all storage pools, regardless of reactor type, could be borated.

References

1 "ENDF/B-V Nuclear Data Guidebook," EPRI-NP 2510, July 1982.

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4 M.D. DeHart and S.M. Bowman, "Validation of the SCALE Broad Structure 44-Group ENDF/B-V Cross Section Library for use in Criticality Safety Analysis," NUREG/CR-6102, Oak Ridge National Laboratory, 1994.

5 O.W. Hermann, et. al., "Validation of the SCALE System for PWR Spent Fuel Isotopic Composition Analyses," ORNL/TM-12667, Oak Ridge National Laboratory, March 1995.

6 W.C. Jordan, et. al., "Validation of KENOV.a Comparison with Critical Experiments," ORNL/CSD/TM-238, Oak Ridge National Laboratory, Oak Ridge National Laboratory, 1986.

7. "Licensing Report for Expanding Storage Capacity in Harris Spent Fuel Pools C and D," HI-971760, Holtec International, May 26, 1998, (Holtec International Proprietary)

Sample Input Deck Listing and
Tables and Figures

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```
=csas26   parm=size=10000000
KENO-VI Input for Storage Cell Calc. High Density Poisoned Rack
238groupndf5  latticecell
'Data From SAS2H - Burned 5 w/o Fuel
o-16      1 0 0.4646E-01 300.00 end
kr-83     1 0 0.3694E-05 300.00 end
rh-103    1 0 0.2639E-04 300.00 end
rh-105    1 0 0.6651E-07 300.00 end
ag-109    1 0 0.4459E-05 300.00 end
xe-131    1 0 0.2215E-04 300.00 end
'xe-135   1 0 0.9315E-08 300.00 end
cs-133    1 0 0.5911E-04 300.00 end
cs-134    1 0 0.5951E-05 300.00 end
cs-135    1 0 0.2129E-04 300.00 end
ba-140    1 0 0.1097E-05 300.00 end
la-140    1 0 0.1485E-06 300.00 end
nd-143    1 0 0.4070E-04 300.00 end
nd-145    1 0 0.3325E-04 300.00 end
pm-147    1 0 0.8045E-05 300.00 end
pm-148    1 0 0.4711E-07 300.00 end
pm-148    1 0 0.6040E-07 300.00 end
pm-149    1 0 0.6407E-07 300.00 end
sm-147    1 0 0.3349E-05 300.00 end
sm-149    1 0 0.1276E-06 300.00 end
sm-150    1 0 0.1409E-04 300.00 end
sm-151    1 0 0.7151E-06 300.00 end
sm-152    1 0 0.5350E-05 300.00 end
eu-153    1 0 0.4698E-05 300.00 end
eu-154    1 0 0.1710E-05 300.00 end
eu-155    1 0 0.6732E-06 300.00 end
gd-154    1 0 0.1215E-06 300.00 end
gd-155    1 0 0.5101E-08 300.00 end
gd-156    1 0 0.2252E-05 300.00 end
gd-157    1 0 0.3928E-08 300.00 end
gd-158    1 0 0.6153E-06 300.00 end
gd-160    1 0 0.3549E-07 300.00 end
u-234     1 0 0.6189E-07 300.00 end
u-235     1 0 0.3502E-03 300.00 end
u-236     1 0 0.1428E-03 300.00 end
u-238     1 0 0.2146E-01 300.00 end
np-237    1 0 0.1383E-04 300.00 end
pu-238    1 0 0.4534E-05 300.00 end
pu-239    1 0 0.1373E-03 300.00 end
pu-240    1 0 0.5351E-04 300.00 end
pu-241    1 0 0.3208E-04 300.00 end
pu-242    1 0 0.1127E-04 300.00 end
am-241    1 0 0.9976E-06 300.00 end
```

```
am-242     1 0 0.2071E-07 300.00 end
am-243     1 0 0.2359E-05 300.00 end
cm-242     1 0 0.3017E-06 300.00 end
cm-244     1 0 0.6846E-06 300.00 end
i-135     1 0 0.2543E-07 300.00 end
'Zirc
cr        2 0 7.5891E-5   300.0 end
fe        2 0 1.4838E-4   300.0 end
zr        2 0 4.2982E-2   300.0 end
'Water w/ 2000 ppm boron
h2o       3 0.99 300.0 end
'b-10      3 0 2.2061E-5   300.0 end
'SS structural material
ss304      4 0.99 300.0 end
'Boral (model as b4c-al using areal density of b-10 @ -- g/cm^2 and 0.18 atom percent b-10 in
nat. b)
'Excluded Proprietary Information
end comp
'squarepitch card excluded - Proprietary Information
more data
dab=999
end more
read param
gen=103 npg=3000 xs1=yes pki=yes gas=yes flx=yes fdn=yes far=yes nb8=999
end param
read geom
'geom cards excluded - Proprietary Information
end geom
read array
ara=1 nux=15 nuy=15 nuz=1 fill
  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
  1  1  2  1  1  2  1  1  1  2  1  1  2  1  1
  1  1  1  1  1  1  1  2  1  1  1  1  1  1  1
  1  1  1  1  2  1  1  1  1  1  2  1  1  1  1
  1  1  2  1  1  1  1  1  1  1  1  1  2  1  1
  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
  1  1  1  2  1  1  1  2  1  1  1  2  1  1  1
  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
  1  1  2  1  1  1  1  1  1  1  1  1  2  1  1
  1  1  1  1  2  1  1  1  1  1  2  1  1  1  1
  1  1  1  1  1  1  1  2  1  1  1  1  1  1  1
  1  1  2  1  1  2  1  1  1  2  1  1  2  1  1
  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
end fill
end array
```

```
read bounds all=mirror end bounds
read mixt sct=2 eps=1.e-01 end mixt
read plot
scr=yes
ttl='w15x15 in High Density Rack'
xul=-11.5 yul= 11.5 zul=0.0
xlr= 11.5 ylr=-11.5 zlr=0.0
uax=1 vdn=-1 nax=750
end plot
end data
end
```

Table 1 Eigenvalue (using infinite multiplication factor) reduction from skipping cells between high reactivity assemblies.

Skipped Cells	PWR	BWR
2	1.03533	1.02628
4	1.01192	1.01503
6	1.00363	1.01218
8	0.99786	1.01059

The initial cause of the loss of cooling could be the failure of a running pump in either the primary or the secondary system, in which case the response required is simply to start the redundant pump. However, it could also be a more significant failure, such as a pipe break or a heat exchanger blockage. To simplify the model, it has been assumed that a repair is necessary. While this is conservative, it is not considered that this unduly biases the conclusions of the overall study.

If the loss of cooling was detected via the control room alarms, the staff has the full 33 hours in which to repair the system. Assuming that it takes at least 16 hours before parts and technical help arrive, then the operator has 17 hours (33 hours less 16 hours) to repair the system. Failure to repair the SFPC system event is modeled as HEP-COOL-REP-E. This case is modeled by fault tree LOC-OCS-U.

what's a realistic estimate

If the loss of cooling was discovered during walkdowns, it has been conservatively assumed the operator has only 9 hours available (allowing 24 hours before ~~loss of cooling was noticed~~). Since it is assumed that it takes at least 16 hours before technical help and parts ~~arrive~~, it is not possible that the SFPC system can be repaired before the bulk boiling ~~would~~ begin. Failure to repair the SFPC system event is modeled as HEP-COOL-REP-L. This case is modeled by fault tree LOC-OCS-L.

4.1.4.2 Relevant Assumptions

- The operators will avoid using raw water (e.g., water not chemically controlled) if possible. Therefore, the operators are assumed to focus solely on restoration of the SFP cooling system in the initial stages of the event

- If the loss of cooling was detected through shift walkdowns, then 24 hours are (conservatively) assumed to have passed before discovery

- It takes 16 hours to contact maintenance personnel, diagnose the cause of failure, and get new parts

- Mean time to repair the SFP cooling system is 10 hours

- Operating staff has received formal training and there are administrative procedures to guide them in initiating repair (NEI commitment no. 8)

- Repair crew is different than the onsite operators

4.1.4.3 Quantification

<u>Human Error Probabilities</u>

The probability of failure to repair SFPC system is represented by the exponential repair model:

$$e^{-\lambda t}$$

where

Risk assessment page A2