Indian Point Nuclear Generating Units 2 and 3
Docket Nos. 50-247/ 50-286-LR

NRC Staff's Response in Opposition to State of New York's Motion for Partial Summary Disposition of NYS Contention 16/16A

Exhibit C

NUREG-1437
Vol. 1

Generic Environmental Impact Statement for License Renewal of Nuclear Plants

Main Report

Final Report

U.S. Nuclear Regulatory Commission

Office of Nuclear Regulatory Research

SAMDA analyses were included in the final environmental impact statements for Limerick 1 and 2 and Comanche Peak 1 and 2 operating license reviews, and the Watts Bar supplemental final environmental statement for operation. These actions are addressed below.

5.4.1.1 Containment Performance

NRC has examined each of five U.S. reactor containment types (BWR Mark I, II and III; PWR Ice Condenser; and PWR Dry) with the purpose of examining the potential failure modes, potential fixes, and the cost benefit of such fixes. This examination has been called the containment performance improvement (CPI) program and has been documented in a series of reports (NUREG/CR-5225; NUREG/CR-5278; NUREG/CR-5528; NUREG/ CR-5529; NUREG/CR-5565; NUREG/CR-5567; NUREG/CR-5575; NUREG/CR-5586; NUREG/CR-5589; NUREG/CR-5602; NUREG/CR-5623; NUREG/ CR-5630). Tables 5.32 through 5.34 summarize the results of this program. As can be seen from these tables, many potential changes were evaluated but only a few containment improvements were identified for site-specific review. The items evaluated in the CPI program were also included in the list of plant-specific SAMDAs examined in the Limerick, Comanche Peak, and Watts Bar FES supplements, discussed later.

5.4.1.2 Individual Plant Examinations

In accordance with NRC's policy statement on severe accidents, each licensee has been requested to perform an individual plant examination (IPE) to look for vulnerabilities to both internal and external initiating events (Generic Letter 88-20, Supplements 1–4). This examination will consider potential improvements on a plant-specific basis. In effect, IPE could be considered equivalent to a monitoring program that looks at the severe accident performance of each licensed plant. Detailed guidance has been issued to each licensee regarding the scope and conduct of IPE and the reporting requirements. NRC staff intends to review each submittal and, if plant modifications not proposed by the licensee appear warranted, to pursue the incorporation of such modifications via NRC's backfit rule (10 CFR Part 50.109). To date, 22 IPEs have been reviewed by NRC. These IPEs have resulted in plant procedural and programmatic improvements (i.e., accident management) and, in only a few cases, minor plant modifications, to further reduce the risk and consequences of severe accidents.

5.4.1.3 Accident Management

Accident management involves the development of procedures that promote the most effective use of available plant equipment and staff in the event of an accident. NRC has indicated its intent (Generic Letter 88-20, Supplement 2) to request that licensees develop an accident management framework that will include implementation of accident management procedures, training, and technical guidance. It is expected that insights gained as a result of IPE will be factored into the accident management program. As discussed earlier, the majority of improvements identified from the completed IPEs to date have been in the area of accident management or other procedural and programmatic improvements.

5.4.1.4 SAMDA Analyses

Site specific SAMDA analyses were performed for Limerick, Comanche Peak, and Watts Bar. A listing of the specific

Table 5.32 Potential boiling-water reactor containment improvements considered in the containment performance improvement program

Number	Potential improvement	Resolution	Comments
1	Enhanced ADS, low pressure water supply, and backup power	Include in IPE	*a*
2	Hardened vent	Implemented for Mark-Is, included in IPE for Mark-II and IIIs	*b*
3	ATWS sized-hardened vent	Drop	*c*
4	External filter	Drop	*c*
5	Dedicated suppression pool cooling	Drop	*c*
6	Alternate decay heat removal	Drop	*c*
7	Core debris control	Drop	*c*
8	Enhanced drywell spray	Drop	*c*
9	Drywell head flood	Drop	*c*
10	Enhanced reactor building DF	Drop	
11	Backup power for hydrogen ignitors (Mark IIIs)	Included in IPE	*d*

Acronyms: ADS = automatic depressurization system, IPE = individual plant examination, ATWS = anticipated transit without scram, DF = decontamination factor.
[a]Analysis showed that potential improvement may be cost beneficial.
[b]Cost beneficial for Mark-Is.
[c]Not cost effective—potential improvement will be too expensive with too little benefit.
[d]May be cost beneficial.

Table 5.33 Potential pressurized-water reactor ice condenser improvements considered in the containment performance improvement program

Potential improvement	Resolution	Comments
Reactor cavity flooding	Drop	Not cost beneficial. Might cause ex-vessel steam explosion.
Backup water to the containment spray system	Drop	Not cost beneficial
Backup power to the air return fan system	Drop	Not cost beneficial. May increase containment pressurization
Reactor depressurization	Include in accident management	Currently being pursued as a viable accident management strategy
Improved hydrogen ignitor system (backup power)	Include in individual plant examination (IPE)	Most cost beneficial of all alternatives considered (although it still does not meet the backfit test). To be looked at within the IPE program
Containment inerting	Drop	Not cost beneficial, may reduce accessibility for maintenance
Filtered vent	Drop	Not cost beneficial
Ex-vessel core debris curb	Drop	Large uncertainty as to effectiveness
Steam generator tube rupture improvements—increased testing	Further research needed	Being examined in separate Nuclear Regulatory Commission program by the Materials Engineering Branch, RES
Containment bypass improvements	Included in generic issues program	Being examined as part of a separate interfacing system loss of coolant accident generic issue (GSI 105)

Table 5.34 Potential pressurized-water reactor (PWR) large, dry containment improvements considered in the containment performance improvement program

Potential improvement	Resolution	Comments
Operator depressurization using power-operated relief valve	Drop	No conclusive findings on its benefit to risk reduction
Addition of a cavity flooding system	Drop	Not cost beneficial. The effect of a flooded cavity on the direct containment heating threats may be beneficial or detrimental, depending on each plant
Addition of hydrogen control system	Assess in individual plant examination (IPE)	Recommend all dry PWR containments assess the likelihood of local hydrogen detonation in the IPE

SAMDAs reviewed for applicability to Limerick is provided in Table 5.35. The staff examined each SAMDA (individually and, in some cases, in combination) to determine its individual risk reduction potential. This risk reduction was then compared with the cost of implementing the SAMDA to provide cost-benefit evidence of its value. Considering that the estimates of risk at Limerick used by the staff in these evaluations were considered to be high and that the uncertainties associated with the costs, effectiveness, and/or operational disadvantages of some SAMDAs were large, the staff concluded that there was no clear evidence that modifications to Limerick were justified for the purpose of further mitigating severe accident risks.

The staff made a similar assessment of SAMDAs for the Comanche Peak Steam Electric Station. A list of the SAMDAs reviewed in this evaluation is provided in Table 5.36. As with the Limerick evaluation, the staff had no basis for concluding that modifications to Comanche Peak were justified for the purpose of further mitigating environmental concerns as they relate to severe accidents. Recently, the staff evaluated SAMDAs for the Watts Bar Nuclear Plant. As in the Limerick and Comanche Peak analyses, no plant modifications were justified for the purpose of further mitigating severe accident risk and consequences.

Several important items from these analyses should be noted.

- First, the SAMDAs considered at Limerick, Comanche Peak, and Watts Bar covered a broad range of accident prevention and mitigation features. These features included the items that were evaluated for all containment types as part of the CPI Program.
- Second, the Limerick analyses were for a plant at a high population site. Since risk to the public is generally proportional to the population surrounding the plant, one would

Table 5.35 Severe accident mitigation design alternatives (SAMDAs) considered for the Limerick Generating Station

1. Installation of alternative means to maintain suppression pool subcooling to improve plant's capability to remove decay heat and prevent containment overpressure challenge

2. Provision of an alternative means of decay heat removal

3a. Installation of containment vent of sufficient size to prevent containment overpressure due to an anticipated transient without scram event

3b. Installation of containment vent and filter of sufficient size to prevent containment overpressure due to an inability to remove decay heat

3c. Installation of containment vent (no filter) of sufficient size to prevent containment overpressure due to an inability to remove decay heat[a]

4. Installation of core debris control devices to prevent core/concrete interaction and remove decay heat from the core debris

5a. Provide enhanced drywell spray capability to increase the reliability for removal of heat from the drywell atmosphere and the core debris, thereby minimizing the threat of containment failure due to overpressure

5b. Provide modification for flooding of the drywell head to help mitigate accidents that result in leakage through the drywell head seal

6. Provide the capability for diesel-driven, low-pressure makeup to the reactor to help in mitigation of core damage resulting from accident sequences in which the reactor vessel is depressurized and all other means of injecting water to the vessel have been lost

7. Improve the reliability of the automatic depressurization system to reduce the probability of vessel failure at high pressure during a severe accident

8. Establish an improved decontamination factor for secondary containment through enhancement to the fire protection system and/or the standby gas treatment system hardware and procedures to improve fission product removal

[a]This SAMDA has been implemented for plants having Mark I containments.

Table 5.36 Listing of severe accident mitigation design alternatives considered for the Comanche Peak Steam Electric Station

1. Additional Instrumentation for Bypass Sequences: Install pressure-monitoring or leak-monitoring instruments (permanent pressure sensors) between the first two pressure isolation valves on low-pressure injection lines, residual heat removal (RHR) suction lines, and high-pressure injection lines. The additional instrumentation would improve the ability to detect valve leakage or open valves, and would decrease the frequency of interfacing system loss-of-coolant accidents (LOCAs).

2. Deliberate Ignition System: Provide a system to promote ignition of combustible gases (hydrogen and carbon monoxide) at low concentrations. The ignition system would prevent large-scale deflagrations or detonations in events involving gradual releases of combustibles (such as from cladding oxidation or core-concrete interactions) but may be ineffective for rapid releases of hydrogen that could occur coincident with reactor vessel failure at high pressure.

3. Reactor Coolant System Depressurization: Provide a capability to rapidly depressurize the reactor coolant system. Reactor depressurization would allow injection using low-pressure systems and would reduce the threat of direct containment heating and induced failures of steam generator tubes and primary coolant piping in the event low-pressure injection systems are not available. Depressurization could be achieved by a system specially designed to manually depressurize the reactor vessel or by actuation of existing pressurizer power-operated relief valves, reactor vessel heat vent valves, and secondary system valves.

4. Independent Containment Spray System: Provide an independent containment spray system, using the existing spray headers if appropriate. The spray system would cool the containment and the core debris, thereby reducing the challenge to containment from overtemperature and long-term overpressure by steam. However, unless the sprays terminate core-concrete interactions, the noncondensable gases released from the concrete are expected to cause the containment to eventually fail by overpressure.

5. Reactor Cavity Flooding System: Provide a capability to flood the reactor cavity before and after reactor vessel breach. Cavity flooding would promote debris coolability, reduce core-concrete interactions and noncondensable gas production, and provide fission product scrubbing.

6. Filtered Containment Venting: Provide a capability to vent the containment through a vent path routed to an external filter. The filtered vent would mitigate challenges to containment from long-term overpressure and hydrogen burn (by reducing the baseline containment pressure) but may not be effective for mitigating energetic events such as hydrogen burns coincident with reactor vessel failure.

Table 5.36 (continued)

7. Additional Diesel Generator: Provide an additional diesel generator with cross-ties to both Class 1E buses. This modification would increase the availability of the AC power system and reduce the frequency of station blackout sequences.

8. Additional DC Battery Capability: Provide additional DC battery capability to ensure eight hours of instrumentation and control power, as opposed to four in the event of a station blackout. This would extend the time available for recovery and reduce the frequency of long-term station blackout sequences.

9. Alternative Means of Core Injection: Provide a capability for makeup water to the reactor using a low-pressure, diesel-driven pump of sufficient capacity and associated piping hardware and procedures. The diesel-driven pump would serve as a backup to the front-line, low-pressure injection systems and could also be used to maintain core cooling in the event of a LOCA.

10. Improved Availability of Recirculation Mode: Provide a system to automatically switch the suction of the safety injection and centrifugal charging pumps to the RHR pump discharge when the refueling water storage tank is depleted. Automatic switchover would reduce the potential for operator error and improve the availability of core cooling in the recirculation mode.

11. Additional Service Water Pump: Add a third 100 percent service water pump to improve the availability of the station service water system. This would reduce the frequency of sequences involving failure of vital plant equipment due to loss of cooling.

generally expect SAMDAs for plants at high population sites to have the most favorable cost-benefit ratio. Since SAMDAs were found not to be justified at Limerick, it is unlikely that they would be justified for plants at other sites.

- Third, plant procedural and programmatic improvements (rather than plant modifications) were the only cost-beneficial improvements identified from these analyses.

5.4.1.5 Conclusion

Although NRC has gained considerable experience regarding severe accident mitigation improvements, the ongoing regulatory programs related to severe accident mitigation (i.e., individual plant examination/individual plant examination of external events and Accident Management) have not been completed for all plants. Since these programs have identified plant programmatic and procedural improvements (and in a few cases, minor plant modification) as cost effective in reducing severe accident consequence and risk, it would be premature to generically conclude that a consideration of severe accident mitigation is not required for license renewal.

However, based on the experiences discussed above, the NRC expects that a

site-specific consideration of severe accident mitigation for license renewal will only identify procedural and programmatic improvements (and perhaps minor hardware changes) as being cost-beneficial in reducing severe accident risk or consequence. Therefore, a site-specific consideration of alternatives to mitigate severe accidents shall be performed for license renewal unless such a consideration has already been included in a previous EIS or related supplement. Staff evaluations of alternatives to mitigate severe accidents have already been completed and included in an EIS or supplement for Limerick, Comanche Peak, and Watts Bar; therefore, severe accident mitigation need not be reassessed for these plants for license renewal.

5.5 SUMMARY AND CONCLUSIONS

The foregoing discussions have dealt with the environmental impacts of accidents during operation after license renewal. The primary assumption for this evaluation is that the frequency (or likelihood of occurrence) of an accident at a given plant would not increase during the plant lifetime (inclusive of the license renewal period) because regulatory controls ensure the plant's licensing basis is maintained and improved, where warranted. However, it was recognized that the changing environment around the plant is not subject to regulatory controls and introduces the potential for changing risk. Estimation of future severe accident consequences and risk was based upon existing risk and consequence analyses found in FES for recently licensed plants because these include severe accident analyses and constitute a representative set of plants and sites for the United States.

5.5.1 Impacts from Design-Basis Accidents

The environmental impacts of postulated accidents were evaluated for the license renewal period in GEIS Chapter 5. All plants have had a previous evaluation of the environmental impacts of design-basis accidents. In addition, the licensee will be required to maintain acceptable design and performance criteria throughout the renewal period. Therefore, the calculated releases from design-basis accidents would not be expected to change. Since the consequences of these events are evaluated for the hypothetical maximally exposed individual at the time of licensing, changes in the plant environment will not affect these evaluations. Therefore, the staff concludes that the environmental impacts of design-basis accidents are of small significance for all plants. Because the environmental impacts of design basis accidents are of small significance and because additional measures to reduce such impacts would be costly, the staff concludes that no mitigation measures beyond those implemented during the current term license would be warranted. This is a Category 1 issue.

5.5.2 Impacts from Severe Accidents

5.5.2.1 Atmospheric Releases

The evaluation of health and dose effects caused by atmospheric releases used a prediction process to identify those plant sites that are bounded by existing analyses. Existing analyses represent only a subset of operating plants. A particular portion of this subset, specifically those plants having severe accident analyses in their respective FESs, was used in this evaluation. EI (which is a function of population and wind direction), in conjunction with the FES severe accident analyses, was then used to develop a means to predict