Task III.D: Radiation Protection

TASK III.D.1: RADIATION SOURCE CONTROL

The objective of this task is to perform evaluations to establish additional design features that should be included in the rulemaking proceeding of Item II.B.8. The purpose of these evaluations is to identify design features that will reduce the potential for exposure to workers at nuclear power plants and to offsite populations following an accident.

ITEM III.D.1.1: PRIMARY COOLANT SOURCES OUTSIDE THE CONTAINMENT STRUCTURE

The three parts of this item are evaluated separately below.

ITEM III.D.1.1(1): REVIEW INFORMATION SUBMITTED BY LICENSEES PERTAINING TO REDUCING LEAKAGE FROM OPERATING SYSTEMS

This item was clarified in NUREG-0737[1] and requirements were issued.

ITEM III.D.1.1(2): REVIEW INFORMATION ON PROVISIONS FOR LEAK DETECTION

DESCRIPTION

This TMI Action Plan[2] item called for evaluations to identify design features that will reduce the potential for radiation exposure to workers at nuclear power plants and to the offsite population following an accident. Item III.D.1.1 called for the staff to evaluate the likelihood of worker exposure and of releases of radioactivity due to potential sources of radiation and airborne radioactivity from primary coolant that may be in systems outside the containment structure following an accident. The adequacy of the existing acceptance criteria for the design of vent-gas and other systems outside the containment structure were to be evaluated and the need for leak detection systems determined. Item III.D.1.1(2) called for NRR to select a contractor to review information on : (1) provisions for leak detection, equipment arrangement drawings, piping drawings, and fabrication criteria (specifications) for systems (e.g., makeup and purification, RHR, RCIC, vent gas, etc.)[3] that may contain substantial amounts of radioactivity after an accident; and (2) primary-to-secondary steam generator leakage. The review was to be performed on selected operating reactors and for plants that were in the OL review stage at the time the TMI-2 accident occurred. Plants were to be selected to provide those typical of each NSSS vendor.

CONCLUSION

Radiation protection of workers has been and continues to be addressed at nuclear power plants by various means, including area radiation monitors, health physics surveys, personnel dosimetry and administrative controls (locked doors to radiation areas, HP procedures, etc.). These provisions have the capability to protect workers from excess exposure both during routine operation and after an accident. In addition, plant systems outside of containment which have the potential to be contaminated with radioactive material (either by leakage from the primary system or by miss operation) are provided with process radiation monitors. These include the following:

PWRs

1. Containment Atmosphere Vent
2. Condenser Air Ejector Steam System

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[1] NUREG-0737, "Clarification of TMI Action Plan Requirements," U.S. Nuclear Regulatory Commission, November 1980, (Supplement 1) January 1983.
[2] NUREG-0660, "NRC Action Plan Developed as a Result of the TMI-2 Accident," U.S. Nuclear Regulatory Commission, May 1980, (Rev. 1) August 1980.
[3] NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations," U.S. Nuclear Regulatory Commission, July 1979.

RHR

1. Containment Cooling Service Water
2. Service Water Waste Processing Effluent (which includes monitoring for the Chemical and Volume Control System)

BWRs

1. Reactor Building Ventilation
2. Condenser Off-Gas
3. RHR Service Water
4. Reactor Building Closed Cooling Water
5. Service Water Radioactive Waste
6. Reactor Water Cleanup

These radiation monitors have the capability to alert the operators if radioactive material is present in the system. Additional radiation monitoring provisions, beyond those discussed above, were addressed in October 1980 when TMI Action Plan[4] Item II.F.1, "Additional Accident-Monitoring Instrumentation," was clarified in NUREG-0737[5] and requirements were issued to licensees. These requirements consisted of the following: (1) noble gas effluent radiological monitors; (2) provisions for continuous sampling of plant effluents for post- accident releases of radioactive iodines and particulates and on-site laboratory capabilities; (3) containment high-range radiation monitor; (4) containment pressure monitor; (5) containment water level monitor; and (6) containment hydrogen concentration monitor. As a result of these requirements, displays and controls were to be added to control rooms for use by operators during normal and abnormal plant conditions. MPAs F-20, F-21, F-22, F-23, F-24, and F-25 were established by the staff to follow licensee implementation of these requirements. Items (1) and (2) are directed toward monitoring releases from the plant and would provide input for taking measures to protect the offsite population, if such measures were necessary.

The advent of Leak-Before-Break (LBB) technology has also focused attention on leakage detection systems and methodologies. On August 28, 1987, the Broad Scope amendment to GDC-4 was published for public comment. This amendment will allow licensees to apply LBB technology to many systems both inside and outside of containment. Stringent acceptance criteria apply, including requirements to have leakage detection systems and/or methodologies in place for both radioactive and non-radioactive systems; the leak detection capability must be equivalent to that specified in Regulatory Guide 1.45.[6] Where LBB is used in the design, such leak detection provisions will provide additional assurance that the release of radioactive material is detected.

Other ongoing activities directly related to the concerns of this item include Issue 66, "Steam Generator Requirements," which addresses primary-to-secondary leakage limits, and Issue 119.5, "Leak Detection Requirements," which addresses leak detection in BWR reactor coolant pressure boundary stainless steel piping (4" diameter or larger) inside or outside of containment.

The safety concerns raised in this issue are similar to those that have been addressed in reactor designs in other issues and in operating practices directed toward worker protection, as outlined above. No additional radiation monitoring, protection, or leak detection provisions have been identified as needed. Therefore, this item was DROPPED from further consideration.

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[4] NUREG-0660, "NRC Action Plan Developed as a Result of the TMI-2 Accident," U.S. Nuclear Regulatory Commission, May 1980, (Rev. 1) August 1980.
[5] NUREG-0737, "Clarification of TMI Action Plan Requirements," U.S. Nuclear Regulatory Commission, November 1980, (Supplement 1) January 1983.
[6] Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," U.S. Nuclear Regulatory Commission, May 1973. [7907100185]

ITEM III.D.1.1(3): DEVELOP PROPOSED SYSTEM ACCEPTANCE CRITERIA

DESCRIPTION

This TMI Action Plan[7] item called for NRR to review the findings of Item III.D.1.1(2), determine the need for requiring leak-detection systems, and develop proposed acceptance criteria for these systems. The proposed acceptance criteria were to be factored into the resolution of Item II.B.8, "Rulemaking Proceeding on Degraded Core Accidents."

CONCLUSION

The need for requiring leak-detection systems and the development of new acceptance criteria for these systems were pursued by the staff in other issues [see Item III.D.1.1(2)]. As a result, Item III.D.1.1(3) did not provide any data for consideration in Item II.B.8 which was resolved in August 1985. Therefore, Item III.D.1.1(3) was DROPPED from further consideration.

ITEM III.D.1.2: RADIOACTIVE GAS MANAGEMENT

DESCRIPTION

Historical Background

The concern expressed in this TMI Action Plan[8] item is that an accident at any operating nuclear plant could result in the release of significant quantities of radioactive noble gases to the containment atmosphere. Since there are no noble gas recovery systems installed in nuclear plants that could process these large volumes of noble gases, there is no viable alternative to eventual discharge of long-lived noble gases to the environment. It was recommended that a study be initiated to determine the applicability and desirability of the use of available technology to minimize the release of radioactive noble gases during and following various postulated accidents. This study should include an investigation of viable alternatives for storage or disposal of the gases, an assessment of the potential pathways for gaseous releases, and consideration of accelerated rates of treatment of large gas volumes present in large containment structures.

Safety Significance

Discharge of large volumes of long-lived noble gases to the environment following an accident can increase the exposure to personnel on site and to the population in close proximity to the site.

Possible Solution

For the purpose of this analysis it is assumed that the study described above will result in increased capacity of the radioactive gas management systems at all plants.

PRIORITY DETERMINATION

Frequency/Consequence Estimate

The magnitude of public risk reduction attributable to increasing the capacity of radioactive gas management systems is not certain, but it is estimated to range from zero to 5%. From calculations for Item III.D.2.1 in NUREG/CR-2800,[9] a 1% decrease in the dose factors for PWR Release Categories 1 through 7 and BWR Release Categories 1 through 4 results in an estimated total public risk reduction of 8,500 man-rem for all plants (144). Assuming a decrease in the dose factors of 0.5% for this issue, the estimated public risk reduction is 4,250 man-rem.

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[7] NUREG-0660, "NRC Action Plan Developed as a Result of the TMI-2 Accident," U.S. Nuclear Regulatory Commission, May 1980, (Rev. 1) August 1980.
[8] NUREG-0660, "NRC Action Plan Developed as a Result of the TMI-2 Accident," U.S. Nuclear Regulatory Commission, May 1980, (Rev. 1) August 1980.
[9] NUREG/CR-2800, "Guidelines for Nuclear Power Plant Safety Issue Prioritization Information Development," U.S. Nuclear Regulatory Commission, February 1983, (Supplement 1) May 1983, (Supplement 2) December 1983, (Supplement 3) September 1985, (Supplement 4) July 1986, (Supplement 5) July 1996.

Cost Estimate

It will be assumed that space is available at all plants for increasing the retention capacity of radioactive gas management systems by installing pressure vessels such as tanks. The hardware cost per plant is estimated to be about $100,000. Engineering and design costs are estimated to be $50,000 and installation costs would be about $100,000. Therefore, the total industry cost for equipment development, installation, support facilities, and construction labor is estimated to be $0.25M per plant and the total industry cost for implementing the possible solution in 144 plants is $36M.

Industry costs for labor and material associated with operation and maintenance of the possible solution are estimated to be similar to those for Item III.D.2.1, i.e., $16M, based on a cost of $4,000/RY.

The NRC cost is assumed to be limited to implementation costs for development and plant installation of increased-capacity gas management systems. It is estimated that 1.5 years (~75 man-weeks) of NRC time would be required for research, criteria development, and engineering development and 0.3 man-week/plant (43 man-weeks) would be required for NRC administrative and technical effort for the review and approval of licensee submittals. Thus, the total NRC cost associated with the possible solution is (75 + 43) man-weeks x $2,000/man-week or $0.24M.

Therefore, the total of all costs related to resolution of this issue is $[36 + 16 + 0.24]M or $52.2M.

Value/Impact Assessment

Based on a total risk reduction of 4,250 man-rem, the value/impact score is given by:

$$S = \frac{4{,}250 \text{ man-rem}}{\$52.2\text{M}}$$

$$= 82 \ \frac{\text{man-rem}}{\$\text{M}}$$

CONCLUSION

An evaluation of this issue was performed by the staff in SECY-81-450[10] in which a portable emergency unit was considered as a viable solution to this issue. This unit could be transported to the site of a serious reactor accident and used to selectively absorb and contain the noble gases from the containment atmosphere. It was concluded in SECY-81-450[11] that the cost of developing and maintaining such a system would be high for a relatively small dose reduction. The evaluation in SECY-81-450[12] supported the value/impact assessment above and the issue was DROPPED from further consideration.

ITEM III.D.1.3: VENTILATION SYSTEM AND RADIOIODINE ADSORBER CRITERIA

The four parts of this item have been combined and analyzed together.

DESCRIPTION

This TMI Action Plan[13] item required the NRC to make provisions to ensure that there is adequate filtration of radioactivity in ventilation exhausts and that acceptable collection efficiencies of radioiodine adsorbers are maintained during accident conditions. Items III.D.1.3(1), (2), and (3) called for various studies and possible modifications/upgrades as well as revisions to several SRP Sections and Regulatory Guides. These items were defined in May 1980 when the TMI Action Plan[14] was published and are now

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[10] SECY-81-450, "Development of a Selective Absorption System Emergency Unit," U.S. Nuclear Regulatory Commission, July 27, 1981. [8108140094]
[11] SECY-81-450, "Development of a Selective Absorption System Emergency Unit," U.S. Nuclear Regulatory Commission, July 27, 1981. [8108140094]
[12] SECY-81-450, "Development of a Selective Absorption System Emergency Unit," U.S. Nuclear Regulatory Commission, July 27, 1981. [8108140094]
[13] NUREG-0660, "NRC Action Plan Developed as a Result of the TMI-2 Accident," U.S. Nuclear Regulatory Commission, May 1980, (Rev. 1) August 1980.
[14] NUREG-0660, "NRC Action Plan Developed as a Result of the TMI-2 Accident," U.S. Nuclear Regulatory Commission, May 1980, (Rev. 1) August 1980.

no longer valid. Subsequent to the publication of the TMI Action Plan, the NRC developed the Severe Accident Research Program (SARP) and the Source Term Program Plan. The objective of Items III.D.1.3(1), (2), and (3) are covered by various programs within the SARP. The Source Term Program provides the mechanism that assures the results of the SARP are incorporated into the licensing process.[15] Item III.D.1.3(4), associated with the evaluation of charcoal adsorber radioiodine collection performance under accident conditions, was completed149 in July 1982 and documented in NUREG/CR-2550.[16]

CONCLUSION

Items III.D.1.3(1), (2), and (3) are covered in the SARP and the Source Term Program Plan and were dropped from further consideration as separate issues; Item III.D.1.3(4) was resolved with no new requirements.

ITEM III.D.1.3(1): DECIDE WHETHER LICENSEES SHOULD PERFORM STUDIES AND MAKE MODIFICATIONS

This item was evaluated in Item III.D.1.3 above and was DROPPED from further consideration.

ITEM III.D.1.3(2): REVIEW AND REVISE SRP

This item was evaluated in Item III.D.1.3 above and was DROPPED from further consideration.

ITEM III.D.1.3(3): REQUIRE LICENSEES TO UPGRADE FILTRATION SYSTEMS

This item was evaluated in Item III.D.1.3 above and was DROPPED from further consideration.

ITEM III.D.1.3(4): SPONSOR STUDIES TO EVALUATE CHARCOAL ADSORBER

This item was evaluated in Item III.D.1.3 above and was determined to be RESOLVED with no new requirements.

ITEM III.D.1.4: RADWASTE SYSTEM DESIGN FEATURES TO AID IN ACCIDENT RECOVERY AND DECONTAMINATION

DESCRIPTION

Historical Background

This TMI Action Plan Item required a study to investigate the improvements that could be made to radwaste system design features to provide the capability to process accident-related liquids and gases and to achieve decontamination effectively.

Safety Significance

The resolution of this issue would have no effect on reducing public risk. Any improvement in radwaste system design features would not reduce the core melt frequency or public dose. However, there is some occupational risk reduction associated with the radwaste system design improvements.

Possible Solutions

For the purpose of this evaluation, it will be assumed that the study will result in the following recommended changes to the radwaste systems of all plants in operation and under construction: (1) piping and connections installed for attaching a portable demineralization system; (2) additional spray nozzles in containment directed for wash-down of major surfaces; and (3) addition of shielding on stairways inside containment.

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[15] Memorandum for J. Funches from R. Mattson, "Comments on Prioritization of Licensing Improvement Issues," February 2, 1983. [8401170099]

[16] NUREG/CR-2550, "Charcoal Performance Under Simulated Accident Conditions," U.S. Nuclear Regulatory Commission, July 1982.

PRIORITY DETERMINATION

Frequency/Consequence Estimate

An analysis of this issue was performed[17] by PNL and it was found that there is no public risk reduction associated with this issue.

Other Considerations

There are several factors to be considered before a conclusion can be drawn on this issue:

1. Accident Occupational Risk Reduction

   The implementation of the possible solution is expected to reduce occupational radiation dose from cleanup, repair, and refurbishment. From studies conducted by PNL in NUREG/CR-2800[18] the total accident occupational risk reduction is 510 man-rem for all 134 affected plants.

2. Implementation Occupational Risk Increase

   Implementation of the solution would require work to be performed in radiation zones while the reactor is in a shut-down mode. Based on an average of 300 man-weeks/plant in a field of 2.5 millirem/hr, the total occupation dose due to implementation back-fit for 71 operating plants was calculated by PNL64 to be 1,630 man-rem.

3. Operation and Maintenance Occupational Risk Increase

   Based on an average time of 1 man-week/plant-yr in a field of 2.5 millirem/hr, the total occupational dose due to operation and maintenance of the possible solution was calculated by PNL64 to be 284 man-rem for all 134 affected plants.

4. Industry Cost Estimate

   PNL has estimated that the total cost for implementation of the solution on all plants is $375M, with operation and maintenance costs amounting to an additional $8.6M. In the event of accidents, industry is expected to save approximately $12M for cleanup, repair, and refurbishment based on a 10% reduction in the occupational dose associated with these accidents. Thus, the net industry cost for this issue is $(375 + 8.6 - 12)M or $372M.

5. NRC Cost Estimate

   NRC costs associated with this issue are insignificant in comparison to industry costs and have been estimated by PNL 64 to be approximately $3M for all plants.

CONCLUSION

A consideration of the risk associated with this issue shows that the occupational dose increases (for implementation and operation and maintenance) of the possible solution far outweigh the occupational risk reduction during accident conditions. In addition, the cost for implementing the solution is very high with no resultant reduction in public dose. As a result, this issue was placed in the DROP category.

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[17] NUREG/CR-2800, "Guidelines for Nuclear Power Plant Safety Issue Prioritization Information Development," U.S. Nuclear Regulatory Commission, February 1983, (Supplement 1) May 1983, (Supplement 2) December 1983, (Supplement 3) September 1985, (Supplement 4) July 1986, (Supplement 5) July 1996.

[18] NUREG/CR-2800, "Guidelines for Nuclear Power Plant Safety Issue Prioritization Information Development," U.S. Nuclear Regulatory Commission, February 1983, (Supplement 1) May 1983, (Supplement 2) December 1983, (Supplement 3) September 1985, (Supplement 4) July 1986, (Supplement 5) July 1996.