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ComEd

October 19, 1995

U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Attention: Document Control Desk

Subject: Dresden Nuclear Power Station Units 2 and 3
Notification of Changes to the Post-Accident Sampling Program
NRC Dockets 50-237 and 50-249

The purpose of this letter is to inform the Nuclear Regulatory Commission of changes to past commitments made in response to NUREG-0737, Item II.B.3, Post-Accident Sampling Capability. Specifically, a number of commitments were made in response to NUREG-0737, Item II.B.3 that have been reviewed and found to be no longer supportive of meeting the objectives of NUREG-0737 and Regulatory Guide 1.97. As a result of these findings, ComEd intends to simplify its post-accident sampling program with the objective of eliminating unnecessary program elements. An existing License Condition to DPR-19 and DPR-25 allows Dresden Station, to change commitments associated with NUREG-0737 using the 50.59 process.

Pursuant to 10 CFR 50.59, ComEd plans to revise section 9.3.2.1 of the Updated Final Safety Analysis Report (UFSAR), of Facility Operating Licenses DPR-19 and DPR-25 to implement changes to the post-accident sampling (PAS) program and the post-accident sampling system (PASS). Eight program changes are planned overall, seven of which have been reviewed previously and accepted by the NRC. The following list briefly describes the program changes for Dresden;

1. The commitment to obtain containment iodine and particulate samples and analyze them for radionuclide constituents will be retracted. During the initial post-TMI time period, this information was expected to be needed for core damage assessment. The site specific core damage procedure in place today is based on a generic procedure developed by the BWR Owners Group and approved by the NRC. The existing core damage assessment procedure does not require this information.

2. The commitment to perform containment atmosphere hydrogen analysis using the PASS will be retracted. Two separate safety related hydrogen monitors required by NUREG-0737, Item II.F.1.6, perform the same function and exceed the requirements of NUREG-0737, Item II.B.3.

3. The commitment to obtain a reactor coolant stripped gas sample and analyze it for radionuclide constituents will be retracted. The existing core damage assessment procedure does not require this information.

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4. The commitment to obtain a reactor coolant stripped gas sample and analyze it to determine dissolved hydrogen concentration will be retracted. This information was originally intended to provide insight on gas buildup within the reactor system that may impact core cooling conditions as well as to provide verification that reactor coolant dissolved oxygen concentration was below 100 ppb. These requirements are not appropriate for BWR plants based on their design and operational characteristics.

5. The commitment to obtain a reactor coolant sample for analysis of dissolved oxygen concentration will be retracted. This information was originally intended to provide information on the corrosion potential for materials and components in contact with the reactor coolant. The critical level for dissolved oxygen concentration was described at 100 ppb. This requirement is not appropriate for BWR plants based on their design and operational characteristics since dissolved oxygen concentrations are not expected to be below 100 ppb.

6. The commitment to obtain a reactor coolant sample for conductivity analysis will be retracted. This capability was originally provided for based on ComEd's initial post-TMI evaluation of reactor coolant chemistry data that might be needed during an accident. This analysis requirement is not described in NUREG-0737 or Regulatory Guide 1.97 and existing programs for core damage assessment do not use this information.

7. The commitment to obtain a sample and analyze it within three hours upon request implied that the request could be made as soon as the accident occurred. The commitment will be modified so that the time allowed to establish sampling/analysis capabilities will be extended to eight hours to obtain a reactor coolant sample for boron analysis and 24 hours for all other samples/analyses (both reactor coolant and containment atmosphere samples). The request to obtain a sample will be made after this period of time and will continue to follow the three hour requirement for obtaining a sample and performing the analysis. This position has been approved for advanced design light water reactors (SECY-93-087) and recently for existing nuclear power plants.

8. The commitment to obtain a reactor coolant sample for pH analysis will be retracted. ComEd's review of this requirement indicates that the information generated is no longer needed or useful during the accident management phase of an accident.

    a. NUREG-0737 does not require pH analysis of reactor coolant for core damage assessment. ComEd's core damage assessment procedures does not require this information.

    b. pH analysis of reactor coolant is a recommendation of Regulatory Guide 1.97 as a Type E variable. Type E variables provide information that can be used to determine the magnitude of a potential release of radioactive materials, specifically radioiodine from the site and for continuously assessing such releases.

c. Emergency procedures in place to estimate the potential radionuclide inventory available for release do not consider water chemistry conditions such as pH. Emergency procedures follow guidelines established by NUREG-1228 for estimating source terms during the accident response period. Factors to be used for quantifying the iodine source term in the gaseous phase available for release are pre-defined and do not require reactor coolant pH as an input. Note that NUREG-1228 was published in 1988, five years after Revision 3 of Regulatory Guide 1.97.

The changes identified in this letter to the PAS program do not in any way decrease the effectiveness of the PAS program in meeting the objective of NUREG-0737, Item II.B.3. ComEd intends to implement these changes by November 1, 1995 using the 50.59 process to revise our NUREG-0737 commitments as provided by the License Condition for Dresden Station.

To the best of my knowledge and belief, the statements contained in this response are true and correct. In some respects, these statements are not based on my personal knowledge, but obtained information furnished by other ComEd employees, contractor employees, and consultants. Such information has been reviewed in accordance with company practice, and I believe it to be reliable.

The proposed FSAR revision is provided as an attachment to this letter. Please direct any questions pertaining to this effort to Joseph Jirka at (708) 663-3837.

Sincerely,

B. Rybak

B. Rybak
Nuclear Licensing Administrator

BR/psb

Attachments:

(1) References
(2) Dresden Nuclear Power Station UFSAR Revision for Section 9.3.2.1

cc: Regional Administrator - RIII
J. Stang, Project Manager, NRR
S. Orth, NRC Region III
C. Vanderniet, Senior Resident Inspector - Dresden
Office of Nuclear Facility Safety - IDNS

Attachment 1

References:

1. NUREG-0737, "Clarification of TMI Action Plan Requirements," USNRC, Nov., 1980, Wash. D.C.

2. Letter to L. O. DelGeorge (CECo) from J. D. Neighbors (NRC), July 22, 1982, "Post Accident Sampling System NUREG-0737, II.B.3 Evaluation Criteria Guidelines."

3. Letter to D. G. Eisenhut (NRC) from T. J. Rausch (CECo), December 29, 1982, "Dresden Station Units 2 and 3, Quad Cities Station Units 1 and 2 Information Concerning NUREG 0737 Item II.B.3, Post Accident Sampling System, NRC Docket Nos. 50-237/249 and 50-254/265"

4. Letter to D. L. Farrar (CECo) from D. B. Vassallo (NRC), April 16, 1984, "Post-Accident Sampling System, NUREG-0737, Item II.B.3," regarding Dresden Units 2 and 3 and Quad Cities Units 1 and 2

5. Letter to H. R. Denton (NRC) from B. Rybak (CECo), July 2, 1984, "Dresden Station Units 2 and 3, Quad Cities Station Units 1 and 2, Post Accident Sampling System TMI, Item II.B.3 Additional Information NRC Docket Nos. 50-237, 50-249, 50-254, and 50-265"

6. Letter to D. L. Farrar (CECo) from J. A. Zwolinski (NRC), January 14, 1985, "NUREG-0737, Item II.B.3 - Post-Accident Sampling System," regarding Dresden Nuclear Power Station, Unit Nos. 2 and 3, Quad Cities Nuclear Power Station, Unit Nos. 1 and 2

7. Letter to D. L. Farrar (CECo) from J. A. Zwolinski (NRC), January 14, 1985, "NUREG-0737, Item II.B.3 - Post-Accident Sampling System," regarding Dresden Nuclear Power Station, Unit Nos. 2 and 3, Quad Cities Nuclear Power Station, Unit Nos. 1 and 2

8. Letter to H. R. Denton (NRC) from B. Rybak (CECo), May 6, 1985, "Dresden Station Units 2 and 3, Quad Cities Station Units 1 and 2 TMI Item II.B.3 - Post Accident Sampling Core Damage Assessment Procedure, NRC Docket Nos. 50-237/249 and 50-254/265"

9. Letter to D. L. Farrar (CECo) from J. A. Zwolinski (NRC), July 23, 1985, "Final Resolution of Criterion 2 (Estimating the Degree of Reactor Core Damage) Relating to the Post-Accident Sampling System," regarding Dresden Nuclear Power Station, Unit Nos. 2 and 3, Quad Cities Station, Unit Nos. 1 and 2

10. Letter to T. E. Murley (NRC) from J. A. Silady (CECo), April 9, 1990, "Dresden Units 2 & 3 Revised Commitments on Post Accident Sampling System, NRC Docket Nos. 50-237 & 249"

Attachment 2

Dresden Nuclear Power Station
UFSAR Revision for Section 9.3.2.1

9.3.2.1 High Radiation Sampling System

The High Radiation Sampling System (HRSS) or Post-Accident Sampling System (PASS) is provided to meet the requirements of NUREG 0737[1] and the recommendations of Regulatory Guide 1.97[2] for the following:

- provide the capability to determine the degree of core damage under degraded core accident conditions through the collection and analysis of reactor coolant and containment atmosphere samples,
- provide for the analysis of reactor coolant to verify the injection of standby liquid control into the reactor system, and
- to assess the corrosion potential of post-accident reactor coolant on components and materials in contact with the coolant.

These requirements/recommendations are met through the installation of the HRSS, and the establishment of a program for the collection and analysis of reactor coolant and containment atmosphere samples and development of a core damage assessment procedure.

9.3.2.1.1 Design Bases

The post-accident sampling (PAS) program provides the capability to sample, transport, and/or analyze reactor coolant and/or containment atmosphere samples from either unit under degraded core accident conditions. The criteria of NUREG 0737, II.B.3 and the recommendations of Regulatory Guide 1.97 were considered in the development of the program. The criteria used for the design and construction of systems to collect post-accident reactor coolant and containment atmosphere samples include the following[3]:

- the capability to promptly obtain for analysis a reactor coolant sample representative of the core area following an accident,
- the capability to obtain a containment atmosphere sample representative of conditions within the containment following an accident,
- the minimization of the volume of reactor coolant and containment atmosphere to be taken from containment during sampling activities,

9.3-1
- the worst case source term as defined by Regulatory Guide 1.3 was used for shielding design and sampling/analysis considerations. The source terms used were as follows;
    - The Reactor coolant source term is based on the release to the coolant of 100% of the noble gas radionuclides, 50% of the halogen radionuclides, and 1% of the particulate radionuclides in an equilibrium reactor core operating at 2561 MWt; and
    - The Containment atmosphere source term is based on the release to the containment of 100% of the noble gas radionuclides and 25% of the halogen radionuclides in an equilibrium core operating at 2561 MWt.

- the capability to obtain and analyze a sample of reactor coolant or containment atmosphere without radiation exposures to any individual exceeding the criteria of General Design Criteria (GDC) 19[4] (Appendix A, 10 CFR Part 50), (i.e. 5 rem. whole body, 75 rem extremities).

9.3-2 The design of the PASS classified the system and components as non-safety-related except where tie-ins are made to a safety-related system. In the latter case, the sample piping up to the first remotely operated isolation valve was classified as safety-related. The sampling system piping and supports were designed to ANSI B31.1. The system components were not designed to seismic Category I requirements but did consider seismic loads due to the potential of routing over safety-related systems. All PASS piping in the reactor building is seismically supported.

9.3.2.1.2 System Description

9.3.2.1.2.1 Program Description

9.3-3 In response to NUREG-0737, Item II.B.3, "Post-Accident Sampling System", the Post-Accident Sampling (PAS) program has been developed. The objectives of the program include the following:

- have the capability to obtain samples, as defined within the site's program, within twenty-four hours from the time the accident begins, with the exception of boron and chloride analysis of reactor coolant,
- have the capability to obtain a reactor coolant sample within eight hours from the time the accident begins for the performance of a boron analysis,
- have the capability to obtain a reactor coolant sample and analyze it for chloride concentration within ninety-six hours from the time that reactor coolant dissolved oxygen concentration is greater than 100 ppb,
- obtain information to be used for the determination of the degree of core damage including the following;
    - reactor coolant radionuclide data including iodines and particulate activity,
    - containment atmosphere radionuclide data including noble gas activity, and
    - containment atmosphere hydrogen concentrations.
- ensure that reactor coolant and containment atmosphere samples can be obtained and analyzed within three hours once the decision has been made to sample.

9.3-4 A core damage procedure has been developed that utilizes industry accepted practices, including the use of radionuclide data to assess the condition of the core during accident conditions.

9.3.2.1.2.1.1 Analytical Program

9.3.2.1.2.1.1.1 Radionuclides

9.3-5 Gamma spectroscopy instrumentation is utilized for the identification of radionuclides in reactor coolant samples, for iodines and particulate, and containment atmosphere samples (LOCA conditions only), for noble gases. Backup systems as well as locations exist in the event that radiological conditions prohibit the use of the instrumentation.

9.3.2.1.2.1.1.2 Hydrogen

9.3-6 The station has two (2) in-line hydrogen monitors for the drywell atmosphere. These monitors meet category I requirements as defined in Regulatory Guide 1.97. One of the monitors is primary, the second is the redundant backup. Hydrogen concentrations in the containment atmosphere are quantified (in percent by volume) via in-line monitoring with these monitors. A description of this system can be found in section 6.2.5.3.2. In addition to functions described in section 6.2.5.3.2, the hydrogen concentration in containment atmosphere is used in the core damage assessment procedure for estimating core damage.

9.3.2.1.2.1.1.3 Chloride

9.3-7 In-line instrumentation is used for chloride analysis of reactor coolant samples. In the event that the in-line instrumentation becomes inoperable and cannot be repaired in time to meet sampling and analysis time requirements, the PAS program provides for backup sampling and laboratory analysis capabilities through grab samples to ensure that sample analyses can be performed within the required time frame.

9.3.2.1.2.1.1.4 Boron

9.3-8 In-line instrumentation is used for boron analysis of reactor coolant samples. In the event that the in-line instrumentation becomes inoperable and cannot be repaired in time to meet sampling and analysis time requirements, the PAS program provides for backup sampling and laboratory analysis capabilities through grab samples to ensure that sample analyses can be performed within the required time frame.

9.3.2.1.2.1.1.5 Dissolved Hydrogen

9.3-9 Reactor coolant dissolved hydrogen analysis is not performed for the following reasons. The core damage assessment procedure does not require this data for assessing the degree of damage to the core. Additionally, the reactor system is designed to remove non-condensible gases during the process of cooling down. Therefore, any dissolved hydrogen concentrations obtained would not be indicative of the hydrogen inventory generated as a result of degrading core conditions. Use of this data would lead to a non-conservative estimate of the degree of core damage.

9.3.2.1.2.1.1.6 Dissolved Oxygen

9.3-10 Reactor coolant dissolved oxygen analysis is not performed for the following reasons. The core damage assessment procedure does not require this data for assessing the degree of damage to the core. The usefulness of this parameter is in the assessment of the corrosion potential of reactor water to materials and components in contact with the coolant. Verification of reactor coolant dissolved oxygen concentrations below 100 ppb is required by NUREG 0737. In a shutdown condition (normal or post-accident), dissolved oxygen concentrations greater than 100 ppb will exist since there is no means available to reduce the concentration to below 100 ppb.

9.3.2.1.2.1.1.7 pH

pH analysis of reactor coolant is a recommendation of Regulatory Guide 1.97 as a Type E variable. The data generated from the analysis can be used to determine the magnitude of a potential release of radioactive materials, specifically radioiodine from the site and for continuously assessing such releases. NUREG-0737 does not require pH analysis of reactor coolant.

9.3-11 During the accident management phase, reactor coolant pH analysis is not performed for several reasons. The core damage assessment procedure does not require this data for assessing the degree of damage to the core and emergency procedures in place to estimate the potential radionuclide inventory available for release do not consider water chemistry conditions such as pH. Emergency procedures follow guidelines established by NUREG-1228[5] for estimating source terms during the accident response period. Factors to be used for quantifying the iodine source term in the gaseous phase available for release are pre-defined and do not require reactor coolant pH as an input. When needed, grab samples of undiluted reactor coolant can be obtained for laboratory analysis during the recovery phase when radiation levels have decreased.

9.3.2.1.2.1.2 Quality Control Program

9.3-12 Station chemistry procedures are in place to ensure the accuracy of the data and the functionality of the system. These procedures ensure that all instrumentation can produce accurate results. Elements of the analytical and radioanalytical instrumentation quality control program include: maintenance, calibration, performance checks and periodic use of the instrumentation. Whenever practical, the instrumentation and/or portions of the PASS system are integrated into normal day to day operational activities to ensure functionality and availability.

9.3-13 The PASS employs a minimum number of valves which will become inaccessible for repairs after an accident. These valves are not within the scope of 10CFR50.49 and are therefore exempt from the requirement for formal documentation. These valves however have been procured to design specifications appropriate for the expected post-accident environmental conditions in which they will operate.

At least once a year the PAS program is tested to verify that the program objectives can be met. In addition, various components of the PASS such as gauges, valves, indicators, switches, and regulators are periodically verified for functionality.

9.3.2.1.2.1.3 Sample Storage and Control

9.3-14 The PAS program ensures that equipment provided for backup sampling is capable of providing at least one sample/day for 7 days following onset of the accident and at least one sample/week until the accident condition no longer exists. A place for storage of these samples has been established onsite and incorporates the use of shielding to minimize the buildup of radiation fields within the immediate area. In some case, the samples may be transported to an off-site facility.

9.3.2.1.2.1.4 Alternative Power Source

9.3-15 A motor control center (MCC) is located in the operating area of the HRSS building and provides a 480-V power supply to the HRSS and HVAC equipment and a 208/120-V power supply for controlling lighting, and heat tracing the sample tubing. This MCC is powered from 480-V bus 26 (unit 2) or bus 36 (unit 3). Should a loss of off-site power event occur, the PAS program provides for an alternative power source that can be linked up. Standby diesel power is available for the HRSS building and the MCC can be energized to meet the time limits for sampling and analysis under post-accident conditions.

9.3.2.1.2.1.5 Radiation Exposure Minimization

9.3-16 The program considers the need to meet GDC 19 requirements. In the development of the program, information regarding the worst case scenario for plant radiation fields during the accident has been used with time motion studies to verify that activities including preparation, sample collection, sample transport, sample analysis, and sample disposal will not result in personnel exposures in excess of GDC 19 requirements.

9.3.2.1.2.2 Post-Accident Sampling System Description

9.3-17 The post-accident sampling system (PASS) provides for the following operational capabilities during post-accident conditions:

- transfer a sample fluid from the source to the sampling area,
- control the temperature and pressure of the sample,
- obtain a post-accident reactor coolant grab sample in a shielded container suitable for transport to an onsite or offsite laboratory for analysis,
- perform in-line chemical analyses of the post-accident reactor coolant sample for chloride, and boron,
- obtain a grab sample of the containment atmosphere in a shielded container suitable for transport to an onsite or offsite laboratory for analysis, and
- store, handle, and return to the plant waste generated by the sampling operations.

9.3.2.1.2.2.1 PASS Components

9.3-18 The PASS consists of systems and equipment needed to safely obtain reactor coolant samples and containment atmosphere samples. The major components of the PASS include the following;

- Liquid Sample Panel (LSP),
- Containment Air Sample Panel (CASP),
- Chemical Analysis Panel (CAP),
- Chemical Monitoring Panel (CMP),
- a cooling rack for thermally hot samples,
- chilled water system,
- a sample waste collection system,
- valves and piping for the HRSS,
- an independent heating, ventilation, air conditioning (HVAC) system,
- controls for the entire system, and
- a communication system to the control room.

P&ID drawings M-1234, sheets 2, 3, and 4, M-1235, M-1236, M-1237, M-1239 sheets 2, 3, and 4, M-1240, M-1241, and M-1242 provide schematic details of the layout of the HRSS.

9.3.2.1.2.2.2 General Arrangement

9.3-19 PASS components for each unit are housed in a dedicated (HRSS) building which is located adjacent to the respective reactor building. A connecting trench extends the piping and electrical lines from the reactor building to the HRSS building.

9.3-20 The HRSS building is designed in accordance with the Uniform Building Code requirements for Zone I. The HRSS buildings for both units are located south of the corresponding reactor buildings. Each building is free standing and arranged into four separate areas as described in the following.

9.3-21 The HRSS building equipment layout is based on dividing the building area into the following four distinct radiation zones;

1. The vestibule area - where preparations are made for entry to the sampling areas. The building is entered through the vestibule area which contains a clothing change area and a portal radiation monitor. The vestibule is separated from the operating area by a wall with a door.

2. The operating area - where all control and sampling manipulations at the panels are performed. The operating area contains the control panels for liquid and containment air sampling, the motor control center, the LSP, the CAP, and the CASP. The HVAC system control panel is located adjacent to the vestibule. An aisle in front of these panels is provided for manual operations such as valve alignment at the panels, calibration, and shielded cask cart movement.

3. The maintenance aisle - which serves as access to the rear of the sampling panels for maintenance purposes. The maintenance aisle behind the sampling panels is separated from the operating area by a combination of concrete shield walls and a shield door.

4. The pit area - which contains the waste tank and pumps, and serves as the pipe and valve gallery. The pit area houses the sample waste tank, the waste pumps, the sample coolers, the chilled water system, and the HRSS building sump. This area is adequately shielded in view of the very high radiation levels associated with post-accident sample wastes that are collected in the waste tank. A 5-foot wide, 3-foot deep concrete trench with removable 2-foot thick concrete covers connects the reactor building and the pit area. Piping carrying process samples, demineralized water, instrument air, electrical power, control cables, and other services are located in the trench.

The interior finishes of the HRSS building are sealed and painted to provide for easy decontamination of wall and floor surfaces. This will provide surfaces which minimize the penetration of any spilled radioactive liquids into the concrete and allow ease of decontamination of areas.

9.3.2.1.2.2.3 <u>High Radiation Sampling System Building Environmental Control</u>

9.3-22 The HRSS building HVAC system provides heating and cooling, filtered and unfiltered exhaust systems, and positive control of airflows. Conditioned air is supplied to the HRSS building to offset the environmental and internal loads seen by the building. A single filter bypass fan is provided for routine operation to prevent the filters from loading. Control of airflows is provided to assure that the HRSS building is maintained at a negative pressure with respect to the environment. The exhaust air flow rate is maintained at approximately 1000 ft^3/min while the intake air flow rate is maintained at an adjustable differential to ensure infiltration into the HRSS building. To control airborne contamination, the building ventilation is designed such that the air flows from the lesser to the higher contaminated zone, i.e., from the vestibule to the operating area to the maintenance aisle and finally to the pit where it is exhausted outside the building to the plants ventilation system. Under normal conditions the exhaust is not filtered. During a post-accident condition, the exhaust would be routed to the exhaust filter unit. For enhanced reliability, redundant exhaust fans are provided on the filtered train. In both cases, the exhaust is routed to the station's 310 foot chimney and is tied into the ventilation duct in the base.

By design, air is exhausted from the three sample panels to control inleakage at approximately 100 ft^3/min for the CAP, 300 ft^3/min for the CASP and 360 ft^3/min for the LSP to control internal leakage. The exhaust air may be passed through a combined high efficiency particulate air (HEPA) and activated carbon filter train. The HRSS building ventilation system is shown in P&ID drawing M-1236 for unit 2 and M-1241 for unit 3.

9.3-23 All components of the HRSS, with the exception of tubing and valves in the reactor building, are located in the HRSS building. The HRSS building temperature is maintained at approximately 75 degrees F. No severe environmental conditions are imposed on the design of the system components. The heating, ventilation, and air conditioning (HVAC) equipment is located outdoors and is designed for -20 to 105 degrees F, and snow and wind loads.

9.3.2.1.2.2.4 Radiation Shielding

9.3-24 The PASS is designed to provide the capability to extract, monitor, analyze, and dispose of samples of reactor coolant and containment atmosphere during post-accident conditions with radiation exposures well below the criteria of General Design Criteria (GDC) 19 (10 CFR 50, Appendix A). To meet GDC 19 requirements, the following criteria were used in the design and construction of the shielding for the HRSS building, the sample panels, and sampling processes:

- limit the dose rate to 15 mrem/hr in general occupancy areas and 100 mrem/hr in areas infrequently occupied except directly in front of the sample panels, and
- limit the whole body exposure to 100 mrem per technician per sampling exercise in the HRSS building.

The HRSS building is provided with 3 foot thick external walls and a 2 foot thick roof to limit the radiation dose inside the building due to the post-accident radiation sources within the reactor building. Within the HRSS building, concrete shield walls protect the technician in the operating area from radiation sources due to sample flow in tubing, panels, and waste collection tanks.

The LSP is provided with a front panel shield consisting of 7 inches of lead shot (0.09 inches in diameter) sandwiched between two ½ inch steel plates. Shield glass viewing ports are provided for observing the sample bottle needle area and the gauges. The integral steel base consists of 5 inches of lead shot (0.09 inches in diameter) sandwiched between two ½ inch steel plates.

Additional radiological protection features include the following;

- The CAP is provided with front panel and base shield similar in size and configuration to the LSP.
- Provisions exist to purge the sample lines in both the LSP and CAP with demineralized water once the sampling and in-line analysis has been completed.
- The CASP has a front panel of 3 inch thick steel plate which provides adequate shielding from radiation fields present within the CASP hardware.
- The sample tubing raceway in the maintenance aisle is provided with a 4 inch thick steel cover to reduce the dose contribution from this source.

To prevent radiation streaming from the gaps around the LSP, CAP, or CASP, these gaps are packed with lead wool. Laboratory procedures and localized shielding are utilized to maintain doses to laboratory workers well below the allowable levels in GDC 19.

9.3.2.1.2.3 PASS Function Descriptions

9.3.2.1.2.3.1 Reactor Coolant Sample Lines

9.3-25 The LSP is designed to obtain samples during degraded core accident conditions from the following sample points:

- the reactor recirculation discharge line of the "B" loop of unit 2 and the "A" loop of unit 3 at a point downstream of the pump discharge isolation valve,
- the low pressure coolant injection (LPCI) discharge header downstream of the containment cooling heat exchangers, and
- downstream of the shutdown cooling (SDC) system heat exchangers.

These sample points ensure that reactor coolant samples can be obtained under the following plant post-accident conditions:

- post-accident with no coolant loss,
- post-accident, ECCS during a small loss-of-coolant accident (LOCA), and
- post-accident, ECCS during a large LOCA.

9.3-26 Reactor coolant sampling during post-accident conditions does not require an isolated auxiliary system (e.g., reactor water cleanup system (RWCU's)) to be placed in operation in order to use the sampling system. The sample lines include containment isolation valves that will close upon initiation of a containment isolation signal or a safety injection signal. These valves can be remotely controlled from the control room to facilitate sampling during and after an accident. P&ID drawings M-26, M-29, M-32, M-357, M-360, and M-363 provide detailed information on the process sample lines.

9.3-27 The sample is transferred from the source to the sampling panel through stainless steel tubing. The sample lines are ½ inch OD Type 304 stainless steel tubing of all welded construction up to the sample panels. Optimum sample velocities have been specified to minimize settling and plateout, and to keep sample lines from clogging.

9.3.2.1.2.3.2 Temperature Control

9.3-28 Cooling of the sample fluid to 120 degrees F has been provided for liquid sample lines having a post-accident temperature greater than 120 degrees F. The cooling is accomplished by shell and tube type heat exchangers.

9.3-29 The sample cooling water is provided by a chilled water system that includes two redundant air-cooled condensing units and direct expansion coils which are immersed in the two chilled water storage tanks (see P&ID drawings M-1237 and M-1242). The chilled water is constantly recirculated and passed through the expansion coils by a set of recirculation pumps. A second set of pumps provide a chilled water supply to the sample cooling rack. The temperature of the chilled water is maintained at 60 degrees F. Thermal storage capacity is provided in the tanks which will allow obtaining at least two high temperature samples even in the event of complete failure of the refrigeration equipment.

9.3.2.1.2.3.3 Containment Atmosphere Sample Lines

9.3-30 The CASP is designed to obtain samples during degraded core accident conditions from the following sample points:

- the drywell atmosphere at three different sample locations: east coolers area, west coolers area, and reactor head vent area,
- the torus atmosphere, and
- the standby gas treatment system.

The sample lines include containment isolation valves that will close upon initiation of a containment isolation signal or a safety injection signal. These valves can be remotely controlled from the control room to facilitate sampling during and after an accident. Flow through the sample line is established through the use of an eductor that uses nitrogen gas. The flow through the sample line is returned to the containment as a means to manage the radioactive gas. P&ID drawing M-1235 and M-1240 provide detailed information on the process sample lines.

9.3-31 The main sample line is a ½ inch OD Type 304 stainless steel tubing tied into existing sample points downstream of the containment isolation valves. To minimize radiation field buildup in the sample line resulting from plateout, the sample tubing is run with large radius bends, flow velocities are maintained at 10 ft/s, and the tubing is heat traced to maintain it at 275 degrees F. To ensure that representative sampling is achieved, welded tubing is used up to the panel. The heat tracing is also needed to prevent condensation from occurring within the sample line.

9.3.2.1.2.3.4 Liquid Sample Panel

9.3-32 The sample extraction takes place in the LSP. The LSP also routes a sample to the chemical analysis panel (CAP) for chemical analysis. The LSP is a free-standing, self-supporting structure containing the necessary sample tubing, valves, and gauges within a totally enclosed panel.

The LSP contains a reactor coolant sampling module that receives the different sources of primary reactor coolant entering, one at a time, at a maximum of 120 degrees F and 1600 operating (2300 design) psig. Design flow rates through the panel are: 1900 cc/min during purging, and 200 cc/min during sampling. The module has power operated valves to automatically stop either purge or sample flow in the event of excessive sample temperatures resulting from failure of the chilled water system.

The LSP has the following capabilities:

- Collection of an undiluted depressurized reactor coolant in a sealed bottle. The bottle is remotely lowered into a shielded cask. The cask is removed from the panel. Depending on the radiation levels, this sample may be analyzed onsite.

- Collection of a diluted (1 to 1000) depressurized sample in a sealed bottle. The bottle is remotely lowered into a shielded cask. The cask is removed from the panel and transported to the onsite laboratory for chemical and isotopic analysis.

- Measurement of chloride, and boron concentrations in a depressurized, undiluted reactor coolant sample which can be routed to the CAP panel.

The reactor coolant sample is drawn from the appropriate sample point after considering plant conditions.

9.3-33 The LSP sample lines are ¼ inch OD Type 304 stainless steel and can be flushed with demineralized water. The purge and flush volumes can be stored in the HRSS waste tank before pumping the wastes to the drywell floor drain sump or to the reactor building equipment drain tank. The design minimizes the potential for leakage of samples. Should a rupture of the reactor coolant line occur anywhere along the sample line outside containment, the containment isolation valves can be remotely closed. The volume of reactor coolant released over time is limited by the sample line size. Leakage within the PASS panels are contained and routed to the HRSS waste tank.

9.3.2.1.2.3.5 Chemical Analysis Panel

9.3-34 The in-line chemical analyses of reactor coolant samples takes place in the CAP which is located next to the LSP and is interconnected with the LSP. The sample input to the CAP is from the LSP where it has been conditioned, i.e., cooled and depressurized to the design requirements of the CAP. The CAP is a free-standing, self-supporting structure containing the necessary valving, tubing, and analytical equipment within a totally enclosed panel. A graphic display showing the sample and support services flow paths, flow and pressure indicators, calibration reagent tanks, and other components are mounted on the front face of the panel. The effluent from the CAP is routed to the waste tank.

To reduce radiation levels, the tubing within the panel is ¼ and ⅛ inch OD Type 304 stainless steel. Provisions have been made for flushing the sample lines with demineralized water.

9.3.2.1.2.3.5.1 Chloride

9.3-35 The CAP provides for in-line determination of chloride in reactor coolant samples. Chloride analysis is performed by an in-line ion chromatograph. Prior to sample analysis, the instrument is checked with an in-line standard. After performance checking, the reactor coolant is routed from the LSP to the CAP. The chloride analysis result is recorded at the CMP. The system is capable of determining chloride concentrations in undiluted samples in the range of 0.1 to 20.0 ppm with an accuracy of approximately $\pm$ 0.05 ppm for concentrations under 0.5 ppm and $\pm$ 20% for concentrations above 1 ppm. An undiluted or diluted sample can also be collected in a shielded cask at the LSP and retained for chloride analysis in the laboratory.

9.3.2.1.2.3.5.2 Boron

9.3-36 The CAP provides for in-line determination of boron in reactor coolant samples. Boron analysis is performed by an in-line ion chromatograph. Prior to sample analysis, the instrument is performance check with a boron standard. After the check, the reactor coolant is routed from the LSP to the CAP. The boron analysis result is recorded at the CMP. The system is capable of determining boron concentrations in undiluted samples in the range of 50 to 1000 ppm with an accuracy of $\pm$ 50 ppm. An undiluted or diluted sample can also be collected in a shielded cask at the LSP and retained for boron analysis in the laboratory.

9.3.2.1.2.3.6 Containment Atmosphere Sample Panel

9.3-37 The containment atmosphere sampling system consists of a control panel with plant valve indications, the containment atmosphere sample (CAS) control panel, the CASP, and the gas partitioner controls. The CASP is an enclosed cabinet with provisions for the connection of a gas partitioner for collecting a sample. The panel encloses a network of valves, tubing (¼ inch OD Type 304 stainless steel), fittings, instruments, and quick-connect couplings. The CASP routes part of the gaseous sample from the main sample line to the gas partitioner. The CAS control panel operates the valves at the CASP and is located in the operating space area. All of the CASP operations, with the exception of the operations mentioned above, are performed in this space. The gas partitioner is designed to separate the iodine and particulate from the noble gas. A gas collection vial is used to capture the noble gas for radionuclide analysis. Gas cylinders required for operation of the CASP are located outside the HRSS building.

9.3.2.1.2.3.7 Control and Monitoring Panels

9.3-38 Three individual control panels for the operation of the LSP, CAP, and CASP are located in the operating area of the HRSS building and shielded from the sample panels by a 3 foot thick concrete wall. Under post-accident conditions, most of the operations for sampling and monitoring are performed from the following panels to limit the radiation dose to the technician from the radioactive fluids in the sample panels.

- PASS control panel - the PASS control panel consists of three sections. Annunciator windows indicating various alarm conditions are located in the top section. The midsection contains a graphic layout displaying the liquid and gaseous sample system flow paths, valves, pumps, and other equipment. All hand switches with indicating lights for operating valves, pumps, and HVAC equipment are located in the lower section of the control panel.

- Chemical monitor panel (CMP) - the CMP is an auxiliary recorder/monitor panel which contains the indicating and recording equipment for the cells and analyzers which are mounted in the CAP. The panel permits the technician to work with, and observe the indicating and recording equipment from a remote location to reduce exposure under post-accident conditions.

- CAS control panel - the CAS control panel contains selector switches, pilot lights, an annunciator system, and a pressure controller and gauge. A mimic diagram of the

CASP flow paths, valves and equipment is also provided on the panel. The technician uses this control panel to select, initiate, and control sample filling exercises.

9.3.2.1.2.3.8 Waste Handling System

The HRSS waste handling system is provided to handle both liquid and gaseous wastes resulting from the sampling operations. In addition to the spent sample itself, waste is generated in the purging of the sample lines. At the conclusion of the sampling sequence, the lines are flushed to reduce the background activity. Each sample extraction produces approximately ten gallons of waste fluid.

9.3-39 The waste handling system consists of a 250-gallon stainless steel collection tank supplied with two horizontal centrifugal discharge pumps which will handle approximately one week of sampling operation. Liquids enter the tank via a 2 inch drain header. The discharge of the tank is directed to the reactor building equipment drain tank during normal operation and may be directed into the drywell floor drain sump during the post-accident mode.

During post-accident conditions, the incoming samples may contain large quantities of dissolved hydrogen which will accumulate in the waste tank. Noble gases dissolved in the sample will also be stripped and will accumulate in the tank. Inerting and evacuation features are provided to control the concentration of these gases. Since the hydrogen concentration can be approximately 30% by volume, the tank is inerted with nitrogen prior to filling to preclude an explosive gas mixture. Since the tank's atmosphere is not monitored, a rupture disc is provided as backup in the event of detonation of a combustible mixture in the tank. For control of gaseous radionuclides, an evacuating compressor will vent the tank's contents back to the drywell. During normal operation, the tank is vented to the HRSS HVAC and operates on a nominal ¼ inch H_2O negative pressure.

9.3.2.1.2.4 Analytical and Radioanalytical Capabilities Description

9.3-40 Analytical and radioanalytical methods used for post-accident sample analyses are reviewed for the following criteria:

- chemical effect of the post-accident coolant matrix,
- time and radiological dose limitations of analyses,
- radiation effect on method and/or equipment,
- compliance with sensitivity and range requirements,
- sample size requirements, and
- accuracy of the analysis methods.

When practical, instrumentation used on a routine basis will be utilized also for fulfilling post-accident analysis requirements. This practice should help to increase the availability and reliability of the method.

9.3.2.1.2.4.1 Facilities

The hot laboratory is located in the chemistry building on the ground floor elevation. A fork lift will be required to transfer the sample cask from the HRSS building (unit 2 or 3) to the chemistry building. The hot laboratory will be used unless radiation fields do not permit the use of the facility. Then backup laboratory instrumentation in the unit 3 HRSS building will be utilized. The area where the samples will be taken for radionuclide analysis is dependent on the radiation fields present in the main counting facilities. Samples will be counted in the counting room adjacent to the hot laboratory unless the radiation fields are greater than 2.5 mR/hr. Then the samples will be taken to counting facilities located in the unit 2 HRSS building where a Post-Accident Radionuclide-Analysis "Portable" system (PARAPS) will be used.

9.3.2.1.2.4.2 Laboratory Instrumentation

9.3-41 Instrumentation and procedures exist onsite to measure for boron concentrations of 0.5 to 3.0 ppm with an accuracy of ± 20%. The analysis is performed on a 1 to 1,000 diluted reactor coolant sample in the hot laboratory or taken to an off-site facility as backup.

9.3-42 Instrumentation and procedures exist onsite to measure for chloride concentrations within a range of 0.1 to 20 ppb with an accuracy of ± 20% for concentrations above 0.5 ppb and ± 0.05 ppb for concentrations under 0.5 ppb. The analysis is performed on a 1 to 1000 diluted reactor coolant sample in the hot laboratory.

9.3.2.1.2.4.3 Radioanalytical Instrumentation

9.3-43 The counting room instrumentation and PARAPS is capable of acquiring nuclear spectra and identifying constituents of the spectrum as well as quantifying each constituent. Both systems consist of a computer system tied in with a germanium detector and are capable of determining radionuclide concentrations in liquids and gases of varying sample sizes and storing this information for future use. The radionuclide analysis capability include provisions to identify and quantify the isotopes of the nuclide categories discussed above to levels corresponding to the source terms given in Regulatory Guide 1.3. Sensitivity of onsite liquid sample analysis capability permits measurement of nuclide concentrations in the range from approximately 1 μCi/g to 10 Ci/g. The PAS program ensures that background levels of radiation in the counting facility are less than 2.5 mR/hr so that the radionuclide analysis will provide results with an acceptably small error (approximately a factor of 2).

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ENDNOTES

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9.3-32 Tracking No. 500120, "System Design Descriptions for Commonwealth Edison Company Dresden Nuclear Station Units 2 and 3," NUS Corp., March 10, 1981, Rev. 0, Document No. 5308-SDD; Tracking No. 258778, Letter to D. G. Eisenhut (NRC) from T. J. Rausch (CECo), December 29, 1982, "Dresden Station Units 2 and 3, Quad Cities Station Units 1 and 2 Information Concerning NUREG 0737 Item II.B.3, Post Accident Sampling System, NRC Docket Nos. 50-237/249 and 50-254/265"; Tracking No. 500120, Letter to D. L. Farrar (CECo) from D. B. Vassallo (NRC), April 16, 1984, "Post-Accident Sampling System, NUREG-0737, Item II.B.3," regarding Dresden Units 2 and 3 and Quad Cities Units 1 and 2.

9.3-33 Tracking No. 500120, "System Design Descriptions for Commonwealth Edison Company Dresden Nuclear Station Units 2 and 3," NUS Corp., March 10, 1981, Rev. 0, Document No. 5308-SDD; UFSAR 9.6.

9.3-34 Tracking No. 500120, "System Design Descriptions for Commonwealth Edison Company Dresden Nuclear Station Units 2 and 3," NUS Corp., March 10, 1981, Rev. 0, Document No. 5308-SDD; Tracking No. 258778, Letter to D. G. Eisenhut (NRC) from T. J. Rausch (CECo), December 29, 1982, "Dresden Station Units 2 and 3, Quad Cities Station Units 1 and 2 Information Concerning NUREG 0737 Item II.B.3, Post Accident Sampling System, NRC Docket Nos. 50-237/249 and 50-254/265"; Tracking No. 500120, Letter to D. L. Farrar (CECo) from D. B. Vassallo (NRC), April 16, 1984, "Post-Accident Sampling System, NUREG-0737, Item II.B.3," regarding Dresden Units 2 and 3 and Quad Cities Units 1 and 2; UFSAR 9.6.

9.3-35 Tracking No. 258778, Letter to D. G. Eisenhut (NRC) from T. J. Rausch (CECo), December 29, 1982, "Dresden Station Units 2 and 3, Quad Cities Station Units 1 and 2 Information Concerning NUREG 0737 Item II.B.3, Post Accident Sampling System, NRC Docket Nos. 50-237/249 and 50-254/265"; Tracking No. 500120, Letter to D. L. Farrar (CECo) from D. B. Vassallo (NRC), April 16, 1984, "Post-Accident Sampling System, NUREG-0737, Item II.B.3," regarding Dresden Units 2 and 3 and Quad Cities Units 1 and 2.

9.3-36 UFSAR 9.6.

9.3-37 Tracking No. 500120, "System Design Descriptions for Commonwealth Edison Company Dresden Nuclear Station Units 2 and 3," NUS Corp., March 10, 1981, Rev. 0, Document No. 5308-SDD; UFSAR 9.6.

9.3-38 Tracking No. 500120, "System Design Descriptions for Commonwealth Edison Company Dresden Nuclear Station Units 2 and 3," NUS Corp., March 10, 1981, Rev. 0, Document No. 5308-SDD; Tracking No. 258778, Letter to D. G. Eisenhut (NRC) from T. J. Rausch (CECo), December 29, 1982, "Dresden Station Units 2 and 3, Quad Cities Station Units 1 and 2 Information Concerning NUREG 0737 Item II.B.3, Post Accident Sampling System, NRC Docket Nos. 50-237/249 and 50-254/265"; Tracking No. 500120, Letter to D. L. Farrar (CECo) from D. B. Vassallo

(NRC), April 16, 1984, "Post-Accident Sampling System, NUREG-0737, Item II.B.3," regarding Dresden Units 2 and 3 and Quad Cities Units 1 and 2.

9.3-39 Tracking No. 500120, "System Design Descriptions for Commonwealth Edison Company Dresden Nuclear Station Units 2 and 3," NUS Corp., March 10, 1981, Rev. 0, Document No. 5308-SDD.

9.3-40 Tracking No. 258778, Letter to D. G. Eisenhut (NRC) from T. J. Rausch (CECo), December 29, 1982, "Dresden Station Units 2 and 3, Quad Cities Station Units 1 and 2 Information Concerning NUREG 0737 Item II.B.3, Post Accident Sampling System, NRC Docket Nos. 50-237/249 and 50-254/265"; Tracking No. 500120, Letter to D. L. Farrar (CECo) from D. B. Vassallo (NRC), April 16, 1984, "Post-Accident Sampling System, NUREG-0737, Item II.B.3," regarding Dresden Units 2 and 3 and Quad Cities Units 1 and 2.

9.3-41 Tracking No. 500120, Letter to H. R. Denton (NRC) from B. Rybak (CECo), September 14, 1984, "Dresden Station Units 2 and 3, Quad Cities Station Units 1 and 2, Post Accident Sampling System TMI, Item II.B.3, Additional Information NRC Docket Nos. 50-237/249 & 254/265"; Tracking No. 500120, Letter to D. L. Farrar (CECo) from J. A. Zwolinski (NRC), January 14, 1985, four additional NUREG 0737 criteria responses reviewed and approved, one criteria left to be resolved.

9.3-42 Tracking No. 258778, Letter to D. G. Eisenhut (NRC) from T. J. Rausch (CECo), December 29, 1982, "Dresden Station Units 2 and 3, Quad Cities Station Units 1 and 2 Information Concerning NUREG 0737 Item II.B.3, Post Accident Sampling System, NRC Docket Nos. 50-237/249 and 50-254/265"; Tracking No. 500120, Letter to D. L. Farrar (CECo) from D. B. Vassallo (NRC), April 16, 1984, "Post-Accident Sampling System, NUREG-0737, Item II.B.3," regarding Dresden Units 2 and 3 and Quad Cities Units 1 and 2.

9.3-43 Tracking No. 258778, Letter to D. G. Eisenhut (NRC) from T. J. Rausch (CECo), December 29, 1982, "Dresden Station Units 2 and 3, Quad Cities Station Units 1 and 2 Information Concerning NUREG 0737 Item II.B.3, Post Accident Sampling System, NRC Docket Nos. 50-237/249 and 50-254/265"; Tracking No. 500120, Letter to D. L. Farrar (CECo) from D. B. Vassallo (NRC), April 16, 1984, "Post-Accident Sampling System, NUREG-0737, Item II.B.3," regarding Dresden Units 2 and 3 and Quad Cities Units 1 and 2.