ATTACHMENT A

Beaver Valley Power Station, Unit No. 2
Proposed Technical Specification Change No. 103

The following is a list of the affected pages:

Affected Pages:	3/4 3-40
	3/4 3-41
	3/4 3-42

NPF-73
add

(Proposed Wording)

TABLE 3.3-6

RADIATION MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	SETPOINT ~~*~~ (3)	MEASUREMENT RANGE	ACTION
1. AREA MONITORS					
a. Fuel Storage Pool Area (2RMF-RQ202)	1	~~*~~ (1)	$\leq$ 75.8 mR/hr	10^{-1} to 10^{4} mR/hr	19
b. Containment Area (2RMR-RQ206 & 207)	2	1, 2, 3 & 4	$\leq$ ~~[illegible].29 x 10^{3}~~ 2.0 x 10^{4} R/hr	1 to 10^{7} R/hr	36
c. Control Room Area (2RMC-RQ201 & 202)	2	1, 2, 3, 4, 5~~##~~ (4) & 6~~##~~ (4)	$\leq$ 0.476 mR/hr	10^{-2} to 10^{3} mR/hr	46, 47
2. PROCESS MONITORS					
a. Containment					
i. Gaseous Activity (Xe-133) RCS Leakage Detection (2RMR-RQ303B)	1	1, 2, 3 & 4	N/A	10^{-6} to 10^{-1} µCi/cc	20
ii. Particulate Activity (I-131) RCS Leakage Detection (2RMR-RQ303A)	1	1, 2, 3 & 4	N/A	10^{-10} to 10^{-5} µCi/cc	20
b. Fuel Building Vent					
i. Gaseous Activity (Xe-133) (2RMF-RQ301B)	1	~~**~~ (2)	$\leq 7.82 \times 10^{-6}$ µCi/cc	10^{-6} to 10^{-1} µCi/cc	21

(1) ~~*~~With fuel in the storage pool or building
(2) ~~**~~With irradiated fuel in the storage pool
(3) ~~#~~Above background
(4) ~~##~~During movement of irradiated fuel

Move to page with TABLE NOTATIONS

(Proposed Wording)

NPF-73 add

TABLE 3.3-6 (Continued)

RADIATION MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	SETPOINT* (3)	MEASUREMENT RANGE	ACTION
2. PROCESS MONITORS (Continued)					
ii. Particulate (I-131) (2RMF-RQ301A)	1	** (2)	$\leq 6.70 \times 10^{-9}$ µCi/cc	10^{-10} to 10^{-5} µCi/cc	21
c. Noble Gas and Effluent Monitors					
i. Supplementary Leak Collection and Release System					
1) Mid Range Noble Gas (Xe-133)(2HVS-RQ109C)	1	1, 2, 3 & 4	N.A.	10^{-4} to 10^{2} µCi/cc	36
2) High Range Noble Gas (Xe-133)(2HVS-RQ109D)	1	1, 2, 3 & 4	N.A.	10^{-1} to 10^{5} µCi/cc	36
ii. Containment Purge Exhaust (Xe-133)(2HVR-RQ104A & B)	1	6	$\leq$ ~~3 x background~~ 1.01×10^{-3} µCi/cc	10^{-6} to 10^{-1} µCi/cc	22
iii. Main Steam Discharge (Kr-88)(2MSS-RQ101A,B & C)	1/SG	1, 2, 3 & 4	$\leq 3.9 \times 10^{-2}$ µCi/cc	10^{-2} to 10^{3} µCi/cc	36

(3) *Above background

move to page with TABLE NOTATIONS

TABLE 3.3-6 (Continued)

ACTION STATEMENTS

TABLE NOTATIONS

ACTION 19 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, perform area surveys of the monitored area with portable monitoring instrumentation at least once per 24 hours.

ACTION 20 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.4.6.1.

ACTION 21 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the applicable ACTION requirements of Specifications 3.9.12 and 3.9.13.

ACTION 22 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.9.9.

ACTION 36 - With the number of OPERABLE channels less than required by the Minimum Channels OPERABLE requirement, either restore the inoperable channel(s) to OPERABLE status within 72 hours, or:

1) Initiate the preplanned alternate method of monitoring the appropriate parameter(s), and

2) Return the channel to OPERABLE status within 30 days, or, explain in the next Semi-Annual Effluent Release Report why the inoperability was not corrected in a timely manner.

ACTION 46 - With the number of OPERABLE channels one less than required by the Minimum Channels OPERABLE requirement, either restore the inoperable channel to OPERABLE status within 7 days or close the control room series normal air intake and exhaust isolation dampers.

ACTION 47 - With no OPERABLE channels either restore one inoperable channel to OPERABLE status within 1 hour or close the control room series normal air intake and exhaust isolation dampers.

(Proposed Wording)

ATTACHMENT B

Beaver Valley Power Station, Unit No. 2
Proposed Technical Specification Change No. 103
REVISION OF TABLE 3.3-6
Radiation Monitor Setpoints

A. DESCRIPTION OF AMENDMENT REQUEST

The proposed amendment will revise the alarm setpoints for the following Beaver Valley Power Station (BVPS) Unit 2 monitors: 2RMR-RQ206 and 207, (in-containment high range area monitors) and 2HVR-RQ104A and B, (containment purge monitors). The change affects Technical Specification 3.3.3.1, Table 3.3-6, Radiation Monitoring Instrumentation. Each of the revised setpoints will be increased.

The proposed amendment also includes editorial changes. Symbols used to designate footnotes were changed to numerical values.

B. BACKGROUND

Monitors RMR-RQ206 and 207 are high range area monitors located in the BVPS Unit 2 reactor containment building. These monitors were installed pursuant to the requirements of Section II.F.1 of NUREG-0737, "Clarification of TMI Action Plan Requirements," which indicate that the monitors must have "the capability to detect and measure the radiation level within the reactor containment during and following an accident." Additionally, NUREG-0578 noted that "The radiation level inside containment is a parameter closely related to the potential for release of radioactive materials in plant effluents." NUREG-0737 also required that technical specifications be submitted.

The existing alarm setpoint value for the in-containment high range area monitors was calculated in 1986 with a basis related to offsite dose consequences at the general emergency classification. In August 1994, the BVPS Emergency Action Levels (EALs) were approved by the Nuclear Regulatory Commission (NRC). These EALs, documented in EPP/I-1, were based on the guidance of NUMARC/NESP-007. The new BVPS EALs use the in-containment high range area monitors as indication of fission product barrier challenges or failures, rather than as indication of effluent releases, as in the past.

Monitors 2HVR-RQ104A and B are purge monitors for the Unit 2 reactor containment building and analyze the ventilation exhaust from the reactor containment building prior to its mixing with other exhaust streams. These monitors serve two functions: (1) to detect abnormal releases and isolate the release in these events, and (2) to alert (via the containment evacuation alarm) refueling personnel of the need to evacuate affected areas so as to maintain exposures as low as reasonably achievable (ALARA). These monitors are in service during operating Modes 5 and 6, and alarm setpoints are required by technical specifications for Mode 6 only.

The existing alarm setpoint value for the containment purge exhaust ventilation monitors was also calculated in 1986 and was based on any significant release of radioactivity, defined as a monitor reading of three times background. The use of this extremely low setpoint value has resulted in past inadvertent engineered safety features (ESF) actuations. The revised setpoint value provides alarm conditions based on offsite dose considerations and as it is a higher value, will provide greater operational flexibility.

C. JUSTIFICATION

The original containment area high range radiation monitors setpoint was based on a release corresponding to the emergency action level established in NUREG-0654, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants." Adverse meteorology was used as well as the design basis loss of coolant accident release model. The high setpoint was based on the general emergency EAL which related to an offsite dose of 1 rem, 5 rem thyroid.

It has now been decided that the containment area high range radiation monitors' process safety limits will be based on the fission product barrier matrix indicators for the fuel clad and containment barriers. The alarm high setpoint is based on the amount of activity dispersed in the containment atmosphere that corresponds to 20% clad failure (2% core inventory). This is consistent with EAL 1.3.5, Significant Radioactivity in Containment. Since this EAL is greater than that specified for the fuel clad barrier, the failure of all three barriers is implicit in this basis. This alarm setpoint will still correspond to a general emergency.

The original containment purge monitors alarm setpoint calculations used two assumptions: First, there would be no significant release, based on the release point effluent monitor safety limits. Second, a source term based on the fuel handling accident source term with 100 hours decay and release of ten percent of the gap activity in the affected fuel assembly. The technical specification limit is for Mode 6. This operational limit was intended to prevent any significant release of radioactivity and was determined to be two times background with the assumption of a clean containment. Thus, the trip setpoint is currently set at three times background.

The current setpoint value has caused numerous inadvertent ESF actuations. Technical Specification Basis 3/4.9.9 states that the purge and exhaust system is necessary to meet the requirements of 10 CFR Part 100 in the event of a fuel handling accident inside containment. Therefore, a revised setpoint value based on offsite dose considerations is being proposed. The high alarm will now correspond to the release concentration that would result in dose rates that are thirty percent of the offsite dose rate limits as documented in the Offsite Dose Calculation Manual (ODCM). The revised setpoint is a higher value which should eliminate the inadvertent ESF actuations associated with the containment purge monitors.

D. SAFETY ANALYSIS

Currently, the in-containment high range area monitors setpoint does not match the BVPS EALs which were approved by the NRC in August 1994. The BVPS EALs were based on the guidance contained in NUMARC/NESP-007, "Methodology for Development of Emergency Action Levels," Rev. 2, 1/92, and NRC Regulatory Guide 1.101, "Emergency Planning and Preparedness for Nuclear Power Reactors," Rev. 3, 8/92. The EALs, to the extent feasible, are based on readily available information such as control room instrumentation readings which, if exceeded, will initiate assessment measures. This information is detailed in the BVPS alarm response procedures, abnormal operating procedures, and emergency operating procedures. Other immediate actions and follow-up actions are identified in the BVPS emergency preparedness plan.

The in-containment high range area monitors will now be used as indicators of fission product barrier challenges or failures. The in-containment high range area monitors are safety related; however, they do not initiate any safety function, nor do they interface with any other safety related system.

The in-containment high range area monitors were designed with the ability for an operator to input radiation level values as alert and high alarm levels, which, upon actuation, create both a visual (lighted icon) and audible alarm in the control room. The proposed change is limited to the high alarm value. Otherwise, the operating and design parameters of the in-containment high range area monitors will not change. These monitors do not provide for any automatic actions of other equipment or systems when an alarm condition occurs.

The current in-containment high range area monitors setpoint value is based on an EAL threshold at the general emergency classification as is the revised setpoint. However, the existing setpoint is based on an offsite dose of 1 rem, 5 rem thyroid. The alarm setpoints corresponded to releases which would lead to a general emergency classification based on offsite doses.

The in-containment high range area monitors setpoints will now be based on the amount of activity dispersed in containment equivalent to twenty percent fuel failure. This is an indicator of the fission product barrier EALs. It is more appropriate that the alarm setpoint be an indicator of fission product barrier degradation. The EAL classification scheme provides for necessary emergency response actions to protect the public. The proposed setpoints are more realistic and would not evacuate the general public unnecessarily.

The Containment Purge and Exhaust System exhausts the containment building and supplies makeup air in operating Modes 5 and 6. The containment purge radiation monitors are located on the containment building exhaust line and provide automatic isolation of the containment in the event of a high alarm. Technical Specification 3/4.9.9 indicates the purpose of the Containment Purge and Exhaust System is to maintain effluent releases within 10 CFR Part 100 requirements during a fuel handling accident inside containment during Mode 6. For purging the containment during Mode 5, the Containment Purge and Exhaust System is capable of directing flow to three flow paths: ventilation vent (the normal path), Supplementary Leak Collection and Release System or the BVPS Unit 1 process vent. Downstream of the containment purge monitors on each of the three release paths is an effluent radiation monitor. These effluent monitors have alarm setpoints, controlled by the ODCM, corresponding to thirty percent of the site 10 CFR Part 50 Appendix I limits. During Mode 6, technical specifications require the discharge to be aligned through the filtered Supplementary Leak Collection and Release System. The proposed technical specification setpoint change is applicable only in Mode 6.

The original containment purge monitors alarm setpoints were intended to prevent any significant release of radioactivity in the event of a fuel handling accident inside containment by isolating the containment purge exhaust on any detectable increase in radioactivity above background levels. This mode of operation was selected at the time of licensing in lieu of performing offsite dose calculations for this accident scenario. (Offsite doses were calculated for a fuel handling accident outside containment as this scenario was deemed more limiting.) While this protocol results in negligible postulated accident releases, the low value of this alarm setpoint has caused numerous inadvertent ESF actuations. Offsite and control room dose calculations recently performed indicate that higher release rates could be tolerated while maintaining postulated offsite and control room doses within a small fraction of those allowed by 10 CFR Part 100 and General Design Criteria 19 (GDC 19), respectively. These analyses were performed assuming release flow through the main filter banks (as required for fuel movement by technical specifications) but without automatic isolation of the containment which is a conservative assumption. The capability for automatic isolation is required during fuel movement.

The value of the containment purge monitors alarm setpoints is based on the ODCM alarm setpoint methodology, with the exception that: (1) the 100-hour decayed fuel handling accident source term was used instead of the ODCM source term; (2) the more restrictive annual average X/Q value for the ventilation vent pathway was used; (3) a flow rate of 7500 cfm (as required by technical specifications during fuel movement) was assumed; and (4) no dilution air flow enters the release path prior or after the containment purge monitors. Numerically, the high alarm setpoint is set to the release rate that would result in offsite dose rates less than thirty percent of the 10 CFR Part 50 Appendix I criteria of whole body dose rate less than 500 mrem/year or skin dose rate less than 3000 mrem/year.

The potential offsite and control room doses will increase as a result of this setpoint change. However, the change is deemed acceptable as: (1) monitor-initiated isolation of the containment release will occur before offsite dose rates exceed 10 CFR Part 50 Appendix I limits, which are a small fraction of 10 CFR Part 100 limits, (2) control room dose will continue to be acceptable with regard to GDC 19, (3) the ODCM is an approved site document, (4) analysis assumptions regarding X/Q, and no filtration are conservative, actual offsite dose rates would be lower, and (5) the separate design basis fuel handling accident dose calculation postulates offsite doses that are a small fraction of 10 CFR Part 100 and control room doses that are within GDC 19 without monitor initiated automatic isolation of the containment purge release.

E. NO SIGNIFICANT HAZARDS EVALUATION

The no significant hazard considerations involved with the proposed amendment have been evaluated, focusing on the three standards set forth in 10 CFR 50.92(c) as quoted below:

> The Commission may make a final determination, pursuant to the procedures in paragraph 50.91, that a proposed amendment to an operating license for a facility licensed under paragraph 50.21(b) or paragraph 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:
>
> (1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or
>
> (2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or
>
> (3) Involve a significant reduction in a margin of safety.

The following evaluation is provided for the no significant hazards consideration standards.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The proposed monitor alarm setpoint changes and editorial changes are administrative in nature. Should the in-containment high range area monitors fail to annunciate or give a false alarm, there would be no effect on any other plant equipment or systems. These monitors are safety related; however, they do not initiate any safety function, nor do they interface with any other safety related system. The monitors' alarm as a visual (lighted icon) and audible alarm in the control room. The operator is then responsible for taking any corrective actions necessary, based on the alarm and Emergency Action Level (EAL) guidelines. The in-containment high range area monitors do not provide for any automatic actions of other equipment or systems when an alarm condition occurs.

   The containment purge monitors are also safety related with the ability for an operator to input a radiation level value for high alarm levels during Mode 6, which upon actuation, create both a visual (lighted icon) and audible alarm in the control room. At the high alarm level, each monitor automatically sends a signal to close the purge supply and exhaust isolation dampers in the containment building. A change in the value of the alarm setpoint has no effect on the performance of the containment purge and exhaust system. The high alarm and subsequent automatic termination of a radioactive release will now be based on offsite dose considerations. There is no credible failure of the monitors associated with a change of the alarm setpoint value.

   The operating and design parameters of the subject radiation monitors will not change. The proposed change affects only the radiation level at which an alarm condition is created and does not affect any accident assumptions. The in-containment high range area monitors' alarm setpoint change will not affect the radiological consequences of an accident. However, since the containment purge monitors revised setpoint is based on offsite dose consequences and is a higher value than the current setpoint of three times the background radiation level, the postulated offsite radiological consequences of a fuel handling accident inside containment would be increased. An analysis of a fuel handling accident inside containment with the purge and exhaust system discharging through the Supplementary Leak Collection and Release System (SLCRS) filter trains was performed and a summary of this analysis is to be added to Chapter 15 of the Updated Final Safety Analysis Report (UFSAR). The analysis which determined the containment purge monitors' setpoint postulated offsite doses that are less than a small fraction (less than twenty-five percent) of the

10 CFR Part 100 guidelines. The fuel handling accident inside containment calculation demonstrated control room operator doses that comply with General Design Criteria (GDC) 19. Therefore, the increased radiological consequences of the change in the alarm setpoint are acceptable. The analysis assumed no isolation, so isolation actuated by the monitor alarm will reduce doses further.

Therefore, the proposed change will not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed radiation monitor alarm revisions cannot initiate a new type of accident. The referenced radiation monitors' alarms cannot initiate a new type of accident, since even a failure of the monitor itself cannot serve as the initiating event of an accident. Operator action is not made solely on a radiation monitor alarm; other plant condition indicators are also evaluated.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the change involve a significant reduction in a margin of safety?

The in-containment high range area monitors have no capability to mitigate the consequences of an accident and do not interface with any safety related system. These monitors are safety related channels which provide indication to the operator of the integrity of the fission product barriers in containment. This indication, combined with other indications of plant conditions may direct an operator to take action to mitigate the consequences of an accident. The alarm setpoint itself does not perform any specific safety related function and the trip value is not referenced in the UFSAR, nor does any site design basis document take credit for this setpoint. Safety limits and limiting safety system settings are not affected by this proposed change. The site will continue to meet the requirements of 10 CFR Part 100 which limits offsite dose following a postulated fission product release.

The containment purge monitors' revised setpoint is based on offsite dose consequences and is a higher value than the current setpoint of three times the background radiation level. Thus the postulated offsite radiological consequences of a fuel handling accident inside containment are increased which reduces the current margin of safety. An analysis of a fuel handling accident inside containment with the purge and exhaust system discharging through the SLCRS filter trains was performed and a summary of this analysis will be added to Chapter 15 of the UFSAR. The analysis postulated offsite doses to be less than twenty-five percent of the 10 CFR Part 100 guidelines and control room operator doses that comply with GDC 19. The analysis shows that the increased radiological consequences of the change in the alarm setpoint are acceptable. Further, the analysis assumed that no isolation would occur; therefore, isolation actuated by the monitors' alarm will reduce the postulated doses.

Therefore, use of the proposed technical specification would not involve a significant reduction in the margin of safety.

F. NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the considerations expressed above, it is concluded that the activities associated with this license amendment request satisfies the no significant hazards consideration standards of 10 CFR 50.92(c) and, accordingly, a no significant hazards consideration finding is justified.

G. UFSAR CHANGES

Suggested UFSAR changes are provided in Attachment C.

ATTACHMENT C

Beaver Valley Power Station, Unit No. 2
Proposed Technical Specification Change No.103

Applicable UFSAR Changes

TABLE 15.0-12

POTENTIAL DOSES DUE TO POSTULATED ACCIDENTS
(Rem)

Postulated Accident	FSAR Section	Exclusion Area Boundary Thyroid	Exclusion Area Boundary Whole Body Gamma	Exclusion Area Boundary Beta Skin	Low Population Zone* Thyroid	Low Population Zone* Whole Body Gamma	Low Population Zone* Beta Skin
Main steam line break	15.1.5						
Pre-accident Iodine spike		10.5	1.2×10^{-2}	4.6×10^{-3}	1.5	1.4×10^{-3}	6.1×10^{-4}
Concurrent Iodine spike		9.1	2.2×10^{-2}	6.7×10^{-3}	3.2	6.8×10^{-3}	2.2×10^{-3}
Loss of nonemergency ac power to the station auxiliaries	15.2.6	1.5×10^{-1}	5.2×10^{-4}	4.1×10^{-4}	2.1×10^{-2}	6.5×10^{-5}	6.8×10^{-5}
Locked rotor	15.3.3	3.25×10^{1}	3.41	2.09	1.44×10^{1}	3.48×10^{-1}	2.17×10^{-1}
Rod ejection	15.4.8						
Containment leakage		4.1×10^{1}	1.9×10^{-1}	6.5×10^{-2}	2.0	9.2×10^{-3}	3.2×10^{-3}
Secondary side		2.2×10^{-1}	5.1×10^{-1}	3.7×10^{-1}	1.1×10^{-2}	2.5×10^{-2}	1.8×10^{-2}
Small line break - loss-of-coolant	15.6.2	1.6×10^{1}	7.0×10^{-2}	2.4×10^{-2}	8.2×10^{-1}	3.4×10^{-3}	1.2×10^{-3}
Steam generator tube rupture	15.6.3						
Pre-accident iodine spike		11.7	6.8×10^{-2}	5.2×10^{-2}	1.0	3.6×10^{-3}	2.7×10^{-3}
Concurrent iodine spike		6.0	7.3×10^{-2}	5.2×10^{-2}	1.0	4.6×10^{-3}	3.0×10^{-3}
Loss-of-coolant	15.6.5						
Containment leakage		2.7×10^{2}	5.3	2.5	1.3×10^{1}	2.6×10^{-1}	1.2×10^{-1}
ECCS leakage		8.3×10^{-1}	1.3×10^{-2}	5.1×10^{-3}	6.3×10^{-1}	1.2×10^{-2}	1.1×10^{-2}
Waste gas system rupture	15.7.1						
Line rupture			3.1×10^{-1}	1.9×10^{-1}			
Tank rupture			1.6×10^{-1}	1.5			
Fuel handling	15.7.4	2.9×10^{1}	2.33	6.58	1.4	1.1×10^{-1}	3.2×10^{-1}
Within Fuel Bldg Within RBC	15.7.4	2.2×10^{1}	1.79	5.07	2.6	2.1×10^{-1}	6.0×10^{-1}

NOTE:

*For duration of accident

— PROPOSED —

15.7.4 Radiological Consequence of Fuel Handling Accidents

15.7.4.1 Identification of Causes and Accident Description

The fuel handling accident is classified as an ANS Condition IV event, faults that are not expected to occur but are postulated because their consequences include the potential for the release of significant amounts of radioactive material.

The fuel handling accident is postulated to occur in the fuel building and in the containment. Environmental release from the containment is precluded by a design which automatically isolates the containment following the detection of radioactivity by the redundant containment purge monitors (Section 11.5).

The fuel handling accident sequence of events consists of the dropping of one fuel assembly on another fuel assembly in the fuel pool, resulting in cladding damage to the fuel rods in the dropped **assembly plus additional rods in the struck assembly with subsequent instantaneous release of all the gap radionuclide inventory.**

The gap radionuclide inventory is based on the minimum time after refueling shutdown of 100 hours and peak inventories for the damaged fuel assemblies. The fuel pool water provides retention capabilities for radioiodines as described in Table 15.7-6.

The radioactivity released from the fuel pool into the fuel building atmosphere is filtered by the supplementary leak collection and release system (Section 6.5.1).

The radioactivity control features of the fuel storage and handling systems inside containment and in the fuel building meet the requirements of GDC 61 (Section 9.1).

15.7.4.2 Analysis of the Effects and Consequences

15.7.4.2.1 Method of Analysis

or reactor containment

The assumptions applied to the evaluation of the release of radioactivity from the fuel and the fuel building are based on Regulatory Guide 1.25, with the exceptions of iodine filter efficiencies which follow the guidance in Regulatory Guide 1.52, and atmospheric dispersion factors, which follow NUREG-0800 (USNRC 1981) (Section 2.3), and I-131 gap activity fraction, which follow NUREG/CR-5009 (USNRC 1988) (Section 3.2.2).

To verify that a fuel handling accident inside containment does not release radioactivity prior to automatic isolation, an evaluation to show that automatic containment isolation occurs upon detection of radioactivity by the redundant containment purge monitors, has been completed. The time required for air to travel from the radiation monitor to the first containment isolation valve is greater than the

— PROPOSED —

15.7.4.3 Radiological Consequences

15.7.4.3.1 Event Within Fuel Buildi

This event ~~A fuel handling accident~~ is defined as the dropping of one spent fuel assembly onto another fuel assembly in the spent fuel storage area. The accident is postulated to cause damage to all of the fuel rods in the dropped assembly plus an additional 50 rods in the struck fuel assembly with subsequent release of all the activity in the fuel rod gap. The gap activity in the core fuel assemblies consists of 10 percent of the core noble gas and iodine activities, except for Kr-85, which is taken as 30 percent of the Kr-85 core activity and I-131, which is taken as 12% of the I-131 core activity at the time of the accident. The damaged fuel assemblies are assumed to have a radial peaking factor of 1.65, ensuring that the analysis addresses the assemblies with the maximum inventory. The gap inventory released into the fuel pool is based on 100 hours of decay resulting from the time between shutdown and movement of the first fuel assembly.

All of the gap activity in the damaged fuel rods is assumed to leak into the fuel pool where 100 percent of the noble gas and 1.0 percent of the iodine is then released into the building. Even though the activity, which leaks into the fuel building atmosphere may be exhausted through filters over a 2-hour period, the analysis is performed assuming the release is instantaneous. The release to the environment occurs at a point on top of the containment, but for accident evaluation the release is considered to be a ground level release.

The radiological consequences of the postulated fuel handling accident within the fuel building are analyzed based on the assumptions listed in Table 15.7-6, with the initial core gap activities given in Table 15.0-7. The resulting releases are shown in Table 15.7-7. Offsite doses are calculated using the preceding releases in combination with the atmospheric dispersion values given in Table 15.0-11 and the methodology described in Appendix 15A.

The radiological consequences of the postulated fuel handling accident in the fuel building, presented in Table 15.0-12, are well within the guidelines of 10 CFR 100, that is, less than 75 Rem thyroid and 6 Rem whole body.

15.7.4.3.2 Event Within Reactor Containment

15.7.5 Spent Fuel Cask Drop Accidents

15.7.5.1 Identification of Causes and Description

Cask handling procedures ensure that a postulated spent fuel cask drop height of 30 feet is not exceeded. If the spent fuel cask trolley limiting devices are removed during cask handling within the plant, the 30-foot drop height is still not exceeded.

— PROPOSED —

<<<<<<<<NEW SECTION>>>>>>>>>

15.7.4.3.2 Event Within Reactor Containment

This event is defined as the dropping of one spent fuel assembly on to a second assembly still in the reactor core. The kinetic energy available is conservatively assumed to be sufficient enough to break a total of 617 fuel rods in 3 fuel assemblies. This is equivalent to breakage of all of the fuel rods in 2.34 assemblies. The assumed scenario is the breaking of 241 fuel rods by compression in the second assembly. The first assembly falls over striking a third assembly. All 264 rods in the first assembly fail due to bending stress. The remaining kinetic energy is dissipated by breaking 112 fuel rods in the third assembly by compression. Since administrative controls limit the movement of spent fuel to one assembly at a time, damage to a spent fuel assembly in the transfer canal or in the transfer tube can only affect one assembly and is therefore not limiting.

The source term assumptions are the same as for the fuel handling accident within the spent fuel building (see 15.7.4.3.1).

All of the gap activity in the damaged fuel rods is assumed to leak into the reactor cavity water where 100 percent of the noble gas and 1.0 percent of the iodine is released instantaneously to the reactor containment building atmosphere. The containment purge and exhaust system (see 6.2.4) would be automatically isolated by high radiation monitor signals, and could be manually isolated by plant operators in response to other indications should automatic isolation be inoperable. For accident evaluation purposes, the release is postulated to continue for 30 days at a rate of 7500 cfm via the containment purge and exhaust system into the supplementary leak collection and release system (SLCRS) (see 6.5.1), where it is filtered prior to release to the environment at a point on top of the containment.

The radiological consequences of the postulated fuel handling accident within the reactor containment are analyzed based on assumptions listed in Table [new A], with the initial core gap activities given in Table 15.0-7. The resulting releases are shown in Table [new-B]. Offsite doses are calculated using the preceding releases in combination with the atmospheric dispersion value given in Table 15.0-11 and the methodology described in Appendix 15A

The radiological consequences of the fuel handling accident in the reactor containment, presented in Table 15.0-12, are a small fraction of the guidelines of 10 CFR 100, that is, less than 75 rem thyroid and 6 rem whole body.

– PROPOSED –

15.7.5.2 Analysis of Effects and Consequences

The details of spent fuel cask handling are provided in Section 9.1.5.

15.7.5.3 Radiological Consequences

Since a spent fuel cask drop exceeding 30 feet cannot occur, no radiological analysis need be performed for a spent fuel cask drop accident.

15.7.6 References for Section 15.7

Underhill, D.W. 1972. Effects of Rupture in a Pressurized Noble Gas Adsorption Bed; Nuclear Safety Volume 13 Number 6.

U.S. Nuclear Regulatory Commission (USNRC 1976). Calculations of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Pressurized Water Reactors (PWR-GALE Code). NUREG-0017.

USNRC 1978. Preparation of Radiological Effluent Technical Specification for Nuclear Power Plants. NUREG-0133.

USNRC 1981. Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants (formerly issued as NUREG 75/087). NUREG-0800.

USNRC 1988. Assessment of the Use of Extended Burnup Fuel in Light Water Power Reactors. NUREG/CR-5009.

DLC 1989. Unit 2 Fuel Handling Accident Doses at EAB, LPZ, Common Control Room. ERS-SFL-89-019.

DLC 1994, Analysis of Radiological Consequences of a Fuel Handling Accident within CNMT at Unit 2 -- EAB, LPZ, and Common Control Room ERS-SFL-92-035

— PROPOSED —

TABLE 15.7-6

(WITHIN FUEL BUILDING)

ASSUMPTIONS USED FOR THE
FUEL HANDLING ACCIDENT ANALYSIS

Power level (MWt)	2,766
Operating time (days)	650
Gap activity	Table 15.0-7
Minimum time since shutdown (hrs)	100
Total number of fuel assemblies in core	157
Number of fuel rods per assembly	264
Fuel damage	1 assembly and 50 fuel rods
Fraction of gap activity released	1.0
Radial peaking factor	1.65
Minimum depth of water between top of the damaged fuel rods and fuel pool surface (ft)	23
Fuel pool decontamination factor	
Iodines	100
Noble gases	1.0
Iodine fraction above pool	
Inorganic	0.75
Organic	0.25
Fuel building filter efficiency (%)	
Inorganic	95
Organic	95
Type of release	puff

— PROPOSED —

TABLE [new-A]

ASSUMPTIONS USED FOR THE FUEL HANDLING ACCIDENT WITHIN CONTAINMENT ANALYSIS

Power Level (Mwt)	2,766
Operating time (days)	650
Gap Activity	Table 15.0-7
Minimum time since shutdown (hrs)	100
Total number of fuel assemblies in core	157
Number of fuel rods per assembly	264
Fuel damage	2.34 assemblies
Fraction of gap activity released	1.0
Radial peaking factor	1.65
Minimum depth of water between the reactor vessel flange and reactor cavity water surface (ft)	23
Reactor cavity water decontamination factor	
Iodines	100
Noble gases	1.0
Iodine fraction above pool	
Inorganic	0.75
Organic	0.25
SLCRS filter efficiency (%)	
Inorganic	95
Organic	95
Release rate (cfm)	7500
Release duration (hrs)	720

– PROPOSED –

TABLE [new-B]

FUEL HANDLING ACCIDENT IN CONTAINMENT RELEASE TO THE ENVIRONMENT

Nuclide	Ci 0-2 hr	Ci 0-720 hr
Kr-83m	1.1E-8	1.6E-8
Kr-85m	3.6E-3	6.5E-3
Kr-85	2.0E+3	5.0E+3
Kr-88	1.0E-6	1.6E-6
Xe-131m	3.9E+2	9.7E+2
Xe-133m	1.6E+3	4.0E+3
Xe-133	1.1E+5	2.6E+5
Xe-135m	2.9E-1	2.9E-1
Xe-135	1.9E+2	4.0E+2
I-131	2.9E+1	7.3E+1
I-132	1.6E+1	2.3E+1
I-133	2.6	6.0
I-135	2.0E-3	3.9E-3

1.6E-8 = 1.6×10^{-8}

-- PROPOSED --

QADMOD

Program QADMOD calculates dose rates at a series of detector locations with shielding for a number of different source points representing volumetric sources. The program is a modified version of the QAD P-5 program written at the Los Alamos Scientific Laboratory by R. E. Malenfant. This program has been upgraded to include: 1) the FASTER geometry routines, 2) a point source option, 3) a translated cylindrical source volume option, and 4) internal library data for conversion factors, build-up factor coefficients, and mass attenuation factors for several materials and compositions.

15A.1 References for Section 15A

DiNunno, J. J.; Anderson, F. D.; Baker, R. E.; and Waterfield, R.L. 1962. Calculation of Distance Factors for Power and Test Reactor Sites, TID 14844.

U.S. Atomic Energy Commission (USAEC) 1974. Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Pressurized Water Reactors. Regulatory Guide 1.4, Revision 2.

INSERTS

— PROPOSED —

INSERT BEFORE 15A.1

TRAILS

Program TRAILS (for Transport of Radioactive mAterial In Linear Systems), developed by Duquesne Light, evaluates activities, dose rates, and integrated doses in the main control room and at offsite locations. Two compartments upstream of the release point can be modeled. Initial activity in each compartment and independent production (e.g., iodine spiking) into compartments can be modeled. Significant parameters can be varied in up to 20 time steps. The dose calculation model within TRAILS is consistent with the semi-infinite submergence and inhalation models suggested by Regulatory Guide 1.4. The gamma dose in the control room is correct for its finite size using the algorithm of Murphy and Campe. Radionuclide progeny buildup due to decay of parent radionuclides is not modeled. TRAILS has been benchmarked against DRAGON 4.

INSERT AT END

DLC 1989, TRAILS: Transport of Radioactive mAterial In Linear Systems, ERS-SFL-89-020

-- PROPOSED --