ENCLOSURE 5 TO SERIAL: HNP-98-128

SHEARON HARRIS NUCLEAR POWER PLANT
DOCKET NO. 50-400/LICENSE NO. NPF-63
REQUEST FOR LICENSE AMENDMENT
TECHNICAL SPECIFICATION CHANGE FOR SPENT FUEL POOL WATER LEVEL AND REVISED FUEL HANDLING ACCIDENT ANALYSES

TECHNICAL SPECIFICATION PAGES

3/4.9.11 WATER LEVEL - NEW AND SPENT FUEL POOLS

LIMITING CONDITION FOR OPERATION

3.9.11 At least 23 feet of water shall be maintained over the top of fuel rods within irradiated fuel assemblies seated in the storage racks.

APPLICABILITY: Whenever irradiated fuel assemblies are in a pool.

ACTION:

a. With the requirements of the above specification not satisfied, suspend all movement of fuel assemblies and crane operations with loads in the affected pool area and restore the water level to within its limit within 4 hours.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.11 At least once per 7 days, when irradiated fuel assemblies are in a pool, the water level in that pool shall be determined to be at least its minimum required depth.

BASES

3/4.9.10 AND 3/4.9.11 WATER LEVEL - REACTOR VESSEL AND NEW AND SPENT FUEL POOLS

The restrictions on minimum water level ensure that sufficient water depth is available to remove 99% of the assumed ~~10%~~ iodine gap activity released from the rupture of an irradiated fuel assembly. The minimum water depth is consistent with the assumptions of the safety analysis.

Insert A

3/4.9.12 FUEL HANDLING BUILDING EMERGENCY EXHAUST SYSTEM

The limitations on the Fuel Handling Building Emergency Exhaust System ensure that all radioactive material released from an irradiated fuel assembly will be filtered through the HEPA filters and charcoal adsorber prior to discharge to the atmosphere. Operation of the system with the heaters operating for at least 10 continuous hours in a 31-day period is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters. The OPERABILITY of this system and the resulting iodine removal capacity are consistent with the assumptions of the safety analyses. ANSI N510-1980 will be used as a procedural guide for surveillance testing. Criteria for laboratory testing of charcoal and for in-place testing of HEPA filters and charcoal adsorbers is based upon removal efficiencies of 95% for organic and elemental forms of radioiodine and 99% for particulate forms. The filter pressure drop was chosen to be half-way between the estimated clean and dirty pressure drops for these components. This assures the full functionality of the filters for a prolonged period, even at the Technical Specification limit.

Insert A

According to Regulatory Guide 1.25, Revision 0, there is 23 feet of water between the top of the damaged fuel bundle and the fuel pool surface during a fuel handling accident. With 23 feet of water, the assumptions of Regulatory Guide 1.25, Revision 0, can be used directly. In practice, this LCO preserves this assumption for the bulk of the fuel in the storage racks. In the case of a single bundle dropped and lying horizontal on top of the spent fuel racks; however, there may be <23 feet of water above the top of the fuel bundle and the surface, indicated by the width of the bundle. To offset this small nonconservatism, the analysis assumes that all fuel rods fail.

3/4.9.11 WATER LEVEL - NEW AND SPENT FUEL POOLS

LIMITING CONDITION FOR OPERATION

3.9.11 At least 23 feet of water shall be maintained over the top of fuel rods within irradiated fuel assemblies seated in the storage racks.

APPLICABILITY: Whenever irradiated fuel assemblies are in a pool.

ACTION:

a. With the requirements of the above specification not satisfied, suspend all movement of fuel assemblies and crane operations with loads in the affected pool area and restore the water level to within its limit within 4 hours.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.11 At least once per 7 days, when irradiated fuel assemblies are in a pool, the water level in that pool shall be determined to be at least its minimum required depth.

3/4.9.10 AND 3/4.9.11 WATER LEVEL - REACTOR VESSEL AND NEW AND SPENT FUEL POOLS

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TECHNICAL SPECIFICATION CHANGE FOR FUEL POOL WATER LEVEL AND REVISED FUEL HANDLING ACCIDENT ANALYSES

HNP FUEL HANDLING ACCIDENT ANALYSIS VS. REG. GUIDE 1.25 ASSUMPTIONS

HNP FUEL HANDLING ACCIDENT ANALYSES Vs REG. GUIDE 1.25 ASSUMPTIONS

Parameter	FHA Analysis Value	Reg. Guide 1.25 Value	
Water Depth - In Containment In FHB	22 feet 21 feet	23 feet	Depths have changed to reflect Harris specific accident configuration.
Iodine Decontamination Factor - In Containment (22 Feet) In FHB (21 Feet)	Inorganic 107 Organic 1 Overall 85 Inorganic 87 Organic 1 Overall 72	Inorganic 133 Organic 1 Overall 100	Decontamination Factors have been revised to reflect the decreased water depth using methodology consistent with Reg. Guide 1.25.
Iodine 131 Release Fraction	12 percent	10 percent	Release fraction has been increased to be consistent with Nureg/CR 5009.
Iodine 129	Included in Source Term	Not Included in Source Term	I-129 was included as this is the only iodine isotope remaining in older BWR assemblies.
Assembly (Core) Average Burnup	40 GWD/MTU	25 GWD/MTU	The burnup is increased to reflect the current Harris core design burnup.
Radial Peaking Factor	1.73	1.65	Peaking factor has been increased to reflect current Harris Tech. Spec.
Charcoal Filter Efficiency for Iodine - In Containment In FHB	90 percent for inorganic 90 percent for organic 95 percent for inorganic 95 percent for organic	90 percent for inorganic 70 percent for organic	The charcoal filter efficiencies are the same as in the current FHA analysis which has been previously reviewed and approved.

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PROPOSED FSAR REVISIONS

15.7.4 Design Basis Fuel Handling Accidents

15.7.4.1 Identification of Causes and Accident Description. The possibility of a fuel handling accident is remote because of the many interlocks, administrative controls, and physical limitations imposed on the fuel handling operations. All refueling operations are conducted in accordance with prescribed procedures under direct surveillance of a senior reactor operator (SRO). The analyzed Fuel Handling Accident inside containment involves dropping a spent fuel assembly resulting in the rupture of the cladding of all the fuel rods (264) in the assembly.

The project worst case Fuel Handling Accident (FHA) in the Fuel Handling Building (FHB) involves dropping a recently discharge (100 hr decayed) PWR assembly (including the handling tool) on top of another recently discharged PWR assembly in a fuel storage rack. The dropped assembly subsequently falls over landing on BWR fuel assemblies in an adjacent storage rack. Fifty fuel rods are projected to fail in the impacted PWR assembly in storage and all of the rods (264) in the dropped assembly fail when the assembly falls over (Reference 15.7.4.5). Due to the upper bail handle of the BWR fuel assemblies extending above the top of the BWR storage racks, up to 52 BWR assemblies could be impacted when the dropped PWR assembly falls over. All of the rods in the impacted BWR assemblies are assumed to fail.

15.7.4.2 Method of Analysis. The following models are postulated for calculation of the fuel handling accidents:

15.7.4.2.1 Fuel handling accident in the Fuel Handling Building (FHB)

1. The accident occurs at 100 hours following the reactor shutdown; i.e., the time at which spent fuel would be first removed from the reactor and moved into the spent fuel pool.

2. The accident results in breakage of all fuel rods in the dropped assembly, 50 rods in the PWR assembly in storage, and all rods in the 52 BWR assemblies in storage.

3. The damaged PWR assemblies are assumed to be the ones operating at the highest power level in the core region being discharged.

4. BWR fuel activities were determined based on minimum decay times allowed for the IF-300 spent fuel cask (Table 15.7.5-1).

5. The power in these assemblies, and corresponding fuel temperatures, establish the total fission product inventory and the fraction of this inventory which is present in the fuel pellet-cladding gap at the time of reactor shutdown.

6. The fuel pellet-cladding gap inventory of fission products is released to the fuel pool water at the time of the accident.

7. The fuel pool water retains a large fraction of the gap activity of halogens by virtue of their solubility by hydrolysis (Table 15.7.4-1). Noble gases are not retained by the water.

8. Other assumption used are as listed in Table 15.7.4-1.

15.7.4.2.2 Fuel handling accident inside containment.

1. The accident occurs at 100 hours following reactor shutdown, i.e., the time at which spent fuel would first be moved from the reactor.

2. The accident results in breakage of all fuel rods in the dropped assembly.

3. The damaged assembly is assumed to be the one operating at the highest power level in the core.

4. The power in this assembly, and corresponding fuel temperatures, establish the total fission product inventory and the fraction of this inventory present in the fuel pellet-cladding gap at the time of reactor shutdown.

5. The fuel pellet-cladding gap inventory of fission products is released to the reactor cavity water at the time of the accident.

6. The reactor cavity water retains a large fraction of the gap activity of halogens by virtue of their solubility by hydrolysis (Table 15.7.4-3). Noble gases are not retained by the water.

7. The accident occurs during refueling with the Containment Purge System in operation.

8. Containment isolation occurs 20 seconds after detection of the accident with resulting filtration.

9. Other assumptions used are listed in Table 15.7.4-3.

15.7.4.3 Radiological Consequences Analysis.

15.7.4.3.1 Postulated fuel handling accident in the FHB. Design of the fuel storage racks and handling facilities in the fuel storage area is such that fuel will always be in a subcritical geometrical array. The design assumes zero boron concentration in the fuel pool water. The spent fuel pool and reactor cavity water contains boron at the refueling boron concentration. Natural convection of the surrounding water provides adequate cooling of the fuel during handling and storage. Cooling of the water is provided by the Spent Fuel Pool Cooling and Cleanup System. At no time during the transfer from the reactor core to the spent fuel storage rack is a fuel assembly removed from the water. Fuel failure during refueling, as a result of inadvertent criticality or overheating, is not possible.

For this evaluation, dropping of a fuel assembly, as described in 15.7.4.1 is assumed to occur, breaching the cladding and releasing the volatile fission products in the gas gap of the fuel pins. In addition to the area radiation monitors located in the Fuel Handling Building (FHB), portable radiation monitors which emit audible alarms are located in this area during fuel handling operations. Doors in the Fuel Handling Building are closed to maintain controlled leakage characteristics in the fuel pool storage region during refueling operations that involve irradiated fuel. Should a fuel assembly be dropped in the fuel transfer canal, or in the spent fuel pool and release radioactivity above a prescribed level, the airborne radiation

monitors will sound an alarm, alerting personnel to the problem. The FHB operating floor area has safety grade, redundant GM tube area monitors at appropriate locations on the FHB walls, monitoring the fuel pool surface and the air volume over the spent and new fuel pools. As described in Section 12.3.4.1.8.3 these GM tube area monitors will detect gross gamma radiation emanating from airborne material breaching the fuel pool surface and drawn up into the FHB ventilation system from a fuel handling accident. When preset levels are reached, a high alarm signal will initiate switchover from normal ventilation to emergency ventilation as described in Section 6.5. The high alarm signal will automatically actuate normal ventilation isolation in a timely manner such that any radioactive material released from the fuel pools in a Fuel Handling Accident (FHA) will not reach the normal ventilation isolation dampers before they are closed.

The FHB normal ventilation exhaust is also monitored by airborne effluent Particulate, Iodine, Gas (PIG) monitor as described in Section 11.5.2.7.2.2. The FHB emergency ventilation exhausts are monitored by a gas monitor as described in Section 11.5.2.7.2.3. The capability to initiate the emergency ventilation system is provided in the control room as described in Section 7.3.1.3.4. The FHB ventilation system switchover time is within an acceptable duration that limits offsite doses to well within 10 CFR 100 limits.

Assumptions and parameters used in evaluating the fuel handling accident outside Containment are as shown in Table 15.7.4-1. Table 15.7.4-9 shows where the assumptions used in the accident evaluation differ from those specified in Regulatory Guide 1.25.

The whole body dose due to immersion and the thyroid dose due to inhalation have been analyzed for the 0-2 hour period at the exclusion area boundary and for the 0-8 hour period at the LPZ outer boundary. The doses calculated at the exclusion area boundary have been evaluated using the 0-2 hour atmospheric dispersion factor, x/Q, and the corresponding doses at the LPZ have been calculated using the 0-8 hour x/Q value. For the purpose of these accidents the activity has been assumed to be instantaneously released from the Fuel Handling Building. The results are listed in Table 15.7.4-2. The resultant doses are well within the guidelines of 10 CFR 100.

15.7.4.3.2 Postulated fuel handling accident inside containment. The possibility of a fuel handling accident inside Containment during refueling is relatively small due to the many physical, administrative, and safety restrictions imposed on refueling operations.

During fuel handling operations, the Containment is kept in an isolable condition, with all ventilation penetrations to the outside atmosphere either closed or capable of being closed on a containment ventilation isolation signal initiated by redundant area radiation monitors. At least one of the two interlock doors on the personnel locks is kept closed. In addition, there are airborne radiation monitors in the Containment and portable monitors with audible alarms located in the fuel handling area during refueling. Should a fuel assembly be dropped and release activity above a prescribed level, the radiation monitors sound an audible alarm, the Containment is isolated and the personnel are evacuated. The containment pre-entry purge lines are automatically closed upon a containment ventilation isolation signal, thus minimizing the escape of any radioactivity.

For analytical purposes, consideration is given to one accident; a drop of a fuel assembly onto the reactor vessel flange by the manipulator crane inside Containment. Assumptions and parameters used in evaluating the fuel handling accident inside Containment are shown in Table 15.7.4-3.

The radiological consequences of a fuel handling accident inside Containment were conservatively evaluated by assuming that containment releases occur for the first 20-second period. It was assumed that further releases were made in a controlled manner through the Reactor Auxiliary Building Filtration System charcoal adsorbers. The assumption of a controlled release was made due to the availability of these charcoal adsorbers and based on calculations showing that containment isolation can be achieved prior to radionuclides reaching the first containment isolation valve.

The activity released inside Containment as a result of a fuel handling accident will be detected by redundant area radiation monitors. The response time for these monitors is expected to be less than 5 seconds when the monitor set point is reduced to 150 mR/hr for refueling operations. Following activity detection, the monitors will initiate the closure of the containment ventilation isolation valves. The valves will require a maximum of 15 seconds to close. The Containment will be isolated in 20 seconds after detection of the accidental release of radioactivity.

The time required for airborne activity to reach the containment isolation valve is based on 1) travel time from the surface of the reactor pool to the nearest intake header and 2) travel time through the duct. The nearest intake header from the pool is located at a distance of 14.8 feet. The average airflow velocity within a distance of less than 3 feet and the velocity at 3 feet from the intake header was estimated using equations from Reference 15.7.4-1. These equations are as follows:

$$V = \frac{Q}{10X^2 + A} \qquad (1)$$

$$V_{av} = \frac{Q}{X\sqrt{10A}} \qquad \tan^{-1} \frac{X\sqrt{10A}}{A} \qquad (2)$$

where,

X = distance outward along axis, ft. (Note: Equation is accurate only when X is less than 1 1/2 D)

V = centerline velocity at distance X from hood, ft/min.

V_{av} = average velocity within a distance X, ft/min.

Q = air flow, cfm

A = area of hood opening, ft^2

D = diameter of round hoods or side of essentially square hoods.

The air velocity beyond 3 feet, though expected to be smaller, is assumed to remain the same as that at 3 feet from the intake header. The size of the intake header is 24" x 24". The average air velocity up to and including 3 feet was estimated at 179.6 ft/min and the average air velocity beyond 3 feet was estimated to be for a given intake rate of 2500 cfm through the header. Therefore the activity would take 27.6 seconds to reach the intake header from the surface of the pool. The travel time in the 26 in. x 30 in. duct (106 ft. in length) would be 3.4 seconds. Therefore the total time required before the activity would reach the isolation valve is 31 seconds.

The assumptions listed in 15.7.4.2.2 and Table 15.7.4-3 are used to calculate the activity releases and offsite doses for the postulated fuel handling accident inside Containment. Table 15.7.4-9 shows where the assumptions used in the accident evaluation differ from those specified in Regulatory Guide 1.25.

The doses from a fuel handling accident occurring inside Containment have been calculated, and have been found to be below the guidelines of 10CFR100. The results of this analysis are presented in Table 15.7.4-4.

15.7.4.4 Deleted.

15.7.4.4.1 Deleted.

15.7.4.4.2 Deleted.

15.7.4.4.3 Deleted.

15.7.4.4.4 Deleted.

15.7.4.5 Deleted.

TABLE 15.7.4-1

PARAMETERS USED IN EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A FUEL HANDLING ACCIDENT INSIDE THE FUEL HANDLING BUILDING

Parameter	Design Basis Assumptions
Source Data:	
Power level, MWt	2830.5
Radial peaking factor	1.73
Core Average Burnup MWD/MTU	40,000
Decay time, hr	100
Number of failed PWR rods	314
Number of failed BWR assemblies	52
Fraction of fission product gases contained in the gap region of the fuel rods, percent	
Kr-85	30
Other Noble Gases	10
Iodine 131	12
Iodine 129	30
Other Iodine	10
Activity Release Data:	
Fraction of gap activity released to pool, percent	100
Minimum water depth above damaged rods, ft.	21
Pool decontamination factor for noble gases	1
Pool decontamination factor for iodine	
Inorganic	87
Organic	1
Overall	72
Iodine chemical form released to fuel building	
Inorganic iodine percent	75
Organic iodine, percent	25

TABLE 15.7.4-1 (Continued)

Parameter	Design Basis Assumptions
Filter Efficiency	
Iodine, inorganic percent	95
Iodine, organic percent	95
Noble gas percent	0
Activity released to atmosphere, (Ci)	
Isotope	
I-129	1.2×10^{-4}
I-131	6.0×10^{1}
I-133	4.9×10^{0}
Xe-131m	5.9×10^{2}
Xe-133	1.1×10^{5}
Xe-133m	1.3×10^{3}
Xe-135	2.1×10^{1}
Kr-85	4.3×10^{4}
Dispersion Data:	
Atmospheric dispersion factors	5 percentile level χ/Qs, (Table 2.3.4-5)
Doses Calculation Models	Reg. Guide 1.25

TABLE 15.7.4-2

RADIOLOGICAL CONSEQUENCES OF A POSTULATED FUEL HANDLING ACCIDENT IN THE FUEL HANDLING BUILDING

Result	Design Basis Assumptions
Exclusion Area Boundary Dose (0 to 2 hr.) (rem)	
Thyroid	1.9×10^{1}
Whole body	8.1×10^{-1}
LPZ Outer Boundary Dose (0 to 8 hrs.) (rem)	
Thyroid	4.4×10^{0}
Whole body	1.8×10^{-1}

TABLE 15.7.4-3

PARAMETERS USED IN EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A FUEL HANDLING ACCIDENT INSIDE CONTAINMENT

Parameter	Design Basis Assumptions
Source Data:	
Power level, MWt	2830.5
Radial peaking factor	1.73
Core average burnup MWD/MTU	40,000
Decay time, hr	100
Number of failed assemblies	1 (264 rods)
Fraction of fission product gases contained in the gap region of the fuel rods, percent	
Kr-85	30
Other Noble Gases	10
Iodine 131	12
Iodine 129	30
Other Iodine	10
Activity Release Data	
Fraction of gap activity released to pool, percent	100
Minimum water depth above damaged rods, ft.	22
Pool decontamination factor for noble gases	1
Pool decontamination factor for iodine	
Inorganic	107
Organic	1
Overall	85
Iodine chemical form released to fuel building	
Inorganic iodine percent	75
Organic iodine, percent	25
Containment Isolation Time Following the Accident (Sec)	20
Containment Pre-Entry Purge (acfm)	37,000
Containment Volume (cu ft.)	2.23×10^6

TABLE 15.7.4-3 (Continued)

Parameter			Design Basis Assumptions
Filter Efficiency			
Iodine, inorganic percent			90
Iodine, organic percent			90
Noble gas percent			0
Activity released to atmosphere, (Ci)			
	Isotope	Before Isolation	Total*
	I-129	6.6×10^{-7}	1.3×10^{-5}
	I-131	4.7	9.0×10^{1}
	I-133	3.9×10^{-1}	7.3
	I-135	2.7×10^{-4}	5.1×10^{-3}
	Xe-131m	2.7	5.0×10^{2}
	Xe-133	5.3×10^{2}	9.5×10^{4}
	Xe-133m	6.0	1.1×10^{3}
	Xe-135	1.0×10^{-1}	1.7×10^{1}
	Kr-85	2.0×10^{1}	3.5×10^{3}
Dispersion Data			
	Atmospheric dispersion factors	5 percentile level χ/Qs, (Table 2.3.4-5)	
	Dose Calculation Models	Reg Guide 1.25	

* Total is the combination of curies before isolation and after through controlled purge.

TABLE 15.7.4-4

RADIOLOGICAL CONSEQUENCES OF A POSTULATED FUEL HANDLING ACCIDENT INSIDE CONTAINMENT

Result	Design Basis Assumptions Before Isolation	Total*
Exclusion Area Boundary Dose (0 to 2 hr.) (rem)		
Thyroid	1.5	2.9×10^{1}
Whole body	4.0×10^{-3}	6.8×10^{-1}
LPZ Outer boundary Dose (0 to 8 hrs.) (rem)		
Thyroid	3.5×10^{-1}	6.6
Whole body	9.1×10^{-4}	1.5×10^{-1}

* Total is the combination of doses before isolation and after isolation through controlled purge.

TABLE 15.7.4-8

FUEL HANDLING ACCIDENT PWR SOURCE TERM

PWR Assembly Source Term at Shutdown T = 0

Isotope	ORIGEN Activity (Ci/MTU)	Assembly Activity (Ci/Ass'y)
I-129	4.25E-02	3.36E-02
I-131	1.09E+06	8.61E+05
I-132	1.56E+06	1.23E+06
I-133	2.09E+06	1.65E+06
I-134	2.39E+06	1.89E+06
I-135	1.86E+06	1.47E+06
Kr-85	1.49E+04	1.18E+04
Kr-85m	2.72E+05	2.15E+05
Kr-87	5.27E+05	4.16E+05
Kr-88	7.67E+05	6.06E+05
Kr-89	9.64E+05	7.62E+05
Xe-131m	8.01E+03	6.33E+03
Xe-133	2.09E+06	1.65E+06
Xe-133m	5.07E+04	4.01E+04
Xe-135	4.39E+05	3.47E+05
Xe-135m	5.65E+05	4.46E+05
Xe-137	2.01E+06	1.59E+06
Xe-138	1.93E+06	1.52E+06

Assembly Peaking Factor = 1.73
Assembly Uranium Loading = 0.4567 MTU

TABLE 15.7.4-9

FUEL HANDLING ACCIDENT ANALYSES VS. REG. GUIDE 1.25 ASSUMPTIONS

Parameter	FHA Analysis Value		Reg. Guide 1.25 Value	
Water Depth				
- In Containment	22 feet		23 feet	
- In FHB	21 feet			
Iodine Decontamination Factor				
- In Containment (22 feet)	Inorganic	107	Inorganic	133
	Organic	1	Organic	1
	Overall	85	Overall	100
- In FHB (21 feet)	Inorganic	87		
	Organic	1		
	Overall	72		
Iodine 131 Release Fraction	12 percent		10 percent	
Iodine 129	Included in Source Term		Not Included in Source Term	
Assembly (Core) Average Burnup	40 GWD/MTU		25 GWD/MTU	
Radial Peaking Factor	1.73		1.65	
Charcoal Filter Efficiency for Iodine				
- In Containment	90 percent for inorganic 90 percent for organic		90 percent for inorganic 70 percent for organic	
- In FHB	95 percent for inorganic 95 percent for organic			