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United States Nuclear Regulatory Commission
Document Control Desk
Washington, D.C. 20555

Perry Nuclear Power Plant
Docket No. 50-440
License Amendment Request: Fuel Handling Accident Reanalysis

Gentlemen:

Amendment of the Facility Operating License (NPF-58) for the Perry Nuclear Power Plant (PNPP) Unit 1 is requested. The proposed changes affect Technical Specification requirements for handling of irradiated fuel in the Containment or the Fuel Handling Building, and certain specifications related to performing core alterations. These changes are based on analysis of the postulated fuel handling and core alteration accidents and transients for PNPP.

On July 20, 1995, the BWR-6 licensees (Grand Gulf Nuclear Station, River Bend Station, Clinton Power Station and PNPP) met with the NRC Staff to discuss the generic aspects of the proposed changes to the Technical Specifications, and the justification for the proposed changes. During this meeting, the NRC Staff was provided with proposed changes to NUREG-1434, BWR-6 Improved Standard Technical Specifications, Rev. 1. The changes requested herein are consistent with the proposed changes to NUREG-1434. In support of the January 1996 PNPP refueling outage schedule, NRC Staff review and approval of this proposal is requested by January 3, 1996.

This proposed amendment is being submitted as part of the cost beneficial licensing action (CBLA) program established within NRR, in which increased priority is granted to requests to reduce requirements that involve high cost without a commensurate safety benefit. The proposed changes were developed to decrease the operational burden placed on refueling outage resources. This change is expected to result in cost reductions in excess of the $100,000 threshold established under the CBLA program, without resulting in a reduction in the regulatory margin of safety.

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Attachment 1 provides a Summary, a Description of Proposed Changes, a Safety Analysis, and an Environmental Consideration. Attachment 2 provides the Significant Hazards Consideration. Attachment 3 provides a copy of the marked-up TS pages, in both the present format and the format required following implementation of Amendment 69 (the improved Technical Specifications). Attachment 4 provides marked-up Bases pages, for information, since Bases are not part of the Technical Specifications, as identified in 10 CFR 50.36(a).

If you have questions or require additional information, please contact Mr. James D. K[illegible]sterman, Manager - Regulatory Affairs at (216) 280-5833.

Very truly yours,

for Donald C. Shelton

BSF:bf

Attachments

cc: NRC Project Manager
NRC Resident Inspector Office
NRC Region III
State of Ohio

I, Robert W. Schrauder, being duly sworn state that (1) I am Director, Perry Nuclear Services Department of the Cleveland Electric Illuminating Company, (2) I am duly authorized to execute and file this certification on behalf of The Cleveland Electric Illuminating Company and Toledo Edison Company, and as the duly authorized agent for Duquesne Light Company, Ohio Edison Company, and Pennsylvania Power Company, and (3) the statements set forth herein are true and correct to the best of my knowledge, information and belief.

Robert W. Schrauder
Robert W. Schrauder

Sworn to and subscribed before me, the 2nd day of November, 1995.

Jane E Mott

MOTT
N... State of Ohio
My Comm... Expires Feb. 20, 2000
(Recorded In Lake County)

SUMMARY

Following reactor shutdown, decay of short-lived fission products greatly reduces the fission product inventory present in irradiated fuel. The Technical Specifications (Limiting Condition for Operation 3.9.4) require only a 24 hour period of reactor subcriticality prior to fuel movement. This short period of time for radioactive decay is an assumption in the fuel handling accident analysis. The changes proposed by this license amendment request are based on a longer decay period and take credit for the reduction in radionuclide inventory available for release in the event of a fuel handling accident.

The proposed changes redefine the fuel handling requirements in two areas, given the longer decay period since the point of subcriticality.

- Requirements associated with INTEGRITY for the Containment and Fuel Handling Building are relaxed (since no credit is taken for these in the new analysis for mitigation of a fuel handling accident).

- Requirements for selected engineered safety feature (ESF) systems (those which are not taken credit for in the new analysis for mitigating a fuel handling accident) are relaxed.

In order to implement the above concepts, the Limiting Conditions for Operation (LCOs) for INTEGRITY and for the selected ESF systems need only apply if fuel which has recently been in the critical reactor core ("recently irradiated fuel") is handled during the first several days of an outage (prior to completion of the longer decay period.) The Bases will be revised to identify recently irradiated fuel as fuel that has occupied part of a critical reactor core within the previous seven days.

In addition, references to CORE ALTERATIONs are deleted from these INTEGRITY and ESF system LCOs. The deletion of the CORE ALTERATION term is justified since a fuel handling accident is the only event during CORE ALTERATIONs that is postulated to result in fuel damage and radiological release, and such fuel handling accidents will be fully enveloped by the proposed APPLICABILITY. The radiological dose analysis that supports this change request shows that, following the longer decay period, containment and treatment of such releases is not necessary to preserve the regulatory safety margins.

The proposed changes do not impact Technical Specification requirements for systems needed to prevent or mitigate CORE ALTERATION events other than the fuel handling accident. They also do not change the requirements for systems needed to mitigate potential vessel draindown events, systems needed for decay heat removal, or the requirements to maintain high water levels over irradiated fuel.

Implementation of the proposed changes will have a significant impact on outage activities at the Perry Nuclear Power Plant (PNPP), reducing outage costs and increasing flexibility with no impact on the margin of safety. Currently, moving large equipment into or out of primary containment (such as chemical-decontamination equipment, inservice examination/test equipment, or large component parts that require repair) must either be completed prior to establishment of INTEGRITY or delayed until after the INTEGRITY period. Real dollar losses are incurred due to inability of specialized contractors to perform their designated activities, and due to delays in performance of "critical path" work.

Additionally, the high level of modification, maintenance, and repair activities inside the containment during outages has in the past resulted in excessive usage on airlock doors, which led to frequent repairs and subsequent test evolutions on the doors. This created the need to limit access through the doors to once every half hour, restricting personnel and equipment processing in and out of the containment area, and resulting in significant losses in productivity.

In the Fuel Handling Building, similar work restraints are experienced during required periods of INTEGRITY, due to limits on equipment and vehicle access. Fuel handling activities need to be stopped to permit such access, which may impact on the outage critical path. Also, productivity losses occur when personnel are involved in multiple evolutions of establishing, maintaining and releasing INTEGRITY.

These factors, coupled with the increased flexibility for scheduling testing and maintenance activities on containment valves and instrumentation, can result in significant accrued cost reductions and productivity enhancements over the remaining operating life of the plant, allowing outage resources to be directed to other activities, which ultimately will result in improvements in plant maintenance, operations and overall safety. The proposed changes will also not impact on the regulatory margin of safety.

DESCRIPTION OF PROPOSED CHANGES

This proposed amendment to the Technical Specifications revises those specifications associated with handling irradiated fuel in the Primary Containment and the Fuel Handling Building, and selected specifications associated with CORE ALTERATIONS. The purpose is to establish a point where OPERABILITY of those systems typically used to mitigate the consequences of a fuel handling accident is no longer required to meet the Standard Review Plan guidance on offsite dose effects (75 rem thyroid, 6 rem whole body).

Specifically, the proposal identifies that only "recently" irradiated fuel contains sufficient fission products to require OPERABILITY of accident mitigation features to meet the accident analysis assumptions. Therefore, the APPLICABILITY requirements for the associated mitigation features are revised.

The following table identifies the PNPP LCOs [Ref. 1] for which the APPLICABILITY is changed from "when handling irradiated fuel assemblies" to "when handling recently irradiated fuel assemblies". The APPLICABILITY of most of these same LCOs is also revised to eliminate the requirement for the LCO to be met during CORE ALTERATIONS. Consistent changes are made to the associated ACTIONS in each of these LCOs, to reflect the changes in the APPLICABILITY. Additionally, following NRC approval of the request, the Bases for these LCOs will be updated to reflect the revised requirements.

Specification Title (Titles are from ITS; thus they may be different than the CTS title)	Improved Tech Spec (ITS) Number	Current Tech Spec (CTS) Number	Add "recently irradiated"?	Delete "Core Altera-tions"?
Primary Containment and Drywell Isolation Instrumentation	3.3.6.1	3.3.2	Yes	Yes
Control Room Emergency Recirculation (CRER) System Instrumentation	3.3.7.1	3.3.7.1	Yes	Yes
Primary Containment Air Locks	3.6.1.2	3.6.1.3	Yes	Yes
Primary Containment Isolation Valves (PCIVs)	3.6.1.3	3.6.4	Yes	Yes
Primary Containment-Shutdown	3.6.1.10	3.6.1.1.2	Yes	Yes
Containment Vacuum Breakers	3.6.1.11	No Change	Yes	Yes
Containment Humidity Control	3.6.1.12	No Change	Yes	Yes
Secondary Containment	3.6.4.1	3.6.6.1	Yes	Yes
Secondary Containment Isolation Valves (SCIVs)	3.6.4.2	N/A	Yes	Yes
Annulus Exhaust Gas Treatment (AEGT) System	3.6.4.3	3.6.6.2	Yes	Yes
Control Room Emergency Recirculation (CRER) System	3.7.3	3.7.2	Yes	Yes
Control Room HVAC System	3.7.4	N/A	Yes	Yes
Fuel Handling Building	3.7.8	3.7.7.2	Yes	N/A
Fuel Handling Building Ventilation Exhaust System	3.7.9	3.7.7.1	Yes	N/A
AC Sources - Shutdown	3.8.2	3.8.1.2	Yes	No
DC Sources - Shutdown	3.8.5	3.8.2.2	Yes	No
Distribution Systems - Shutdown	3.8.8	3.8.3.2	Yes	No

In addition, Notes in the Primary Containment-Shutdown Specifications (CTS and ITS) are deleted. These Notes deal with an allowance previously granted by Amendment 35, which permits several vent and drain line pathways to exist (breaching Containment) provided that the plant had been subcritical for at least seven (7) days. These notes are no longer necessary, based on the changes discussed above.

SAFETY ANALYSIS

Summary of Original Licensing Basis

The fuel handling accident in the Fuel Handling Building is evaluated in the PNPP Updated Safety Analysis Report (USAR) [Ref. 2] Section 15.7.4. The design basis analysis is based on the Standard Review Plan (SRP) [Ref. 3] Section 15.7.4 and Regulatory Guide (RG) 1.25 [Ref. 4]. The limiting event is the drop of an irradiated channeled fuel assembly onto unchanneled stored spent fuel bundles. The conservative analysis determined that 108 rods fail as a result of the impact. The radioactive release is filtered by the fuel handling area ventilation exhaust system. The Technical Specifications define OPERABILITY requirements and surveillance intervals for the fuel handling area ventilation exhaust radiation monitors, the fuel handling area exhaust system, and the associated electrical power systems. In the original design basis calculations for this accident, these systems are credited for limiting the transport of fission products to the environment. The resultant radiological effects at the Site Boundary are 0.989 rem whole body and 6.59 rem thyroid (see USAR Table 15.7-20).

The original analysis for the fuel handling accident in the Containment is presented in USAR Section 15.7.6. The limiting event is the drop of an irradiated channeled fuel assembly onto the reactor core. The conservative analysis determined that 124 rods fail as a result of the impacts. The analysis assumes that the Containment Vessel and Drywell Purge System is running, and the radioactive release causes high radiation signals to isolate this flow path after 20 seconds. This 20 second release is treated as an instantaneous release to the environment. The remainder of the release is assumed to be released to the environs (via the Annulus Exhaust Gas Treatment System (AEGTS)), with assumptions identical to those presented in USAR Section 15.6.5 for the LOCA. The radiological effects at the Site Boundary are 0.597 rem whole body and 65.4 rem thyroid (see USAR Table 15.7-27).

For the fuel handling accidents described above, Containment and Fuel Handling Building integrity and the associated AEGT or fuel handling area ventilation systems work together to minimize the dose effect of the fission products released from the fuel. The associated radiological consequences are held to less than the SRP 15.7.4 guidance of "well within" the 10CFR100.11 limits. The SRP further defines "well within" for the fuel handling accidents to be 25% of the 10CFR100 limits (specifically, 75 rem thyroid and 6 rem whole body). Because these systems are credited in the original design basis fuel handling accident analysis for mitigating the release of radioactive material, appropriate operating restrictions are imposed by the Technical Specifications.

Reanalysis of Fuel Handling Accident

Reanalysis of the radiological release portion of the fuel handling accident for PNPP has been performed to evaluate the impact of not crediting Containment and Fuel Handling Building INTEGRITY and various engineered safety feature (ESF) systems that are currently used to reduce the consequences of the analyzed events (i.e., the reanalysis assumes an instantaneous release to the environment with no holdup or treatment).

The radiological reanalysis evaluated only the Fuel Handling Accident in the Containment, since it is the bounding analysis. This is simply because more rods are assumed to fail in the Containment accident than in the Fuel Handling Building accident (124 versus 108), and, since no credit is taken for INTEGRITY or ESF mitigation, there is no difference in the subsequent dispersion of the releases. Precursors to the events and the sequence of the events are unchanged from that described in the USAR up to and including the point when a fraction of the radioactivity is released from the surface of the pool water. From this point, the analysis was expanded to evaluate the effects of various decay time periods beyond 24 hours, in conjunction with the assumption that those systems normally used to mitigate the accident are not utilized to minimize the dose effects.

The analysis approach was to calculate the length of decay period necessary to keep calculated whole body and thyroid dose below the Standard Review Plan (SRP) guidance levels (75 rem thyroid, 6 rem whole body). The analysis results have previously been submitted to the NRC for review (see letter dated March 16, 1990 [Ref. 5]), and independent calculations have already been performed by the NRC staff which confirmed the results (see the NRC Safety Evaluation for PNPP Operating License Amendment No. 35, dated September 28, 1990.)

The analysis demonstrated that for the worst case drop, the regulatory dose guidance of SRP 15.7.4 is satisfied without credit for building INTEGRITY or the ESF systems discussed above, provided a decay period longer than 24 hours is assumed.

Key assumptions used in the analysis are as follows (note that these assumptions are identical to those utilized in the March 16, 1990 letter; no new or different offsite dose analyses were performed for this current license amendment request).

- A decay period of seven days (168 hours) subcriticality was assumed.
- Regulatory Guide 1.25 [Ref. 6] assumptions are followed with the exception that the releases are treated as instantaneous releases rather than taking credit for holdup delays of up to two hours. This highly conservative assumption assumes that there is no containment barrier of any kind, no dilution prior to the release, and no filtering done by any ventilation system.
- Current USAR information is used for impact energy and its absorption by the fuel assemblies, and for the number of resultant fuel rod failures.
- PNPP short term (accident) atmospheric dispersion (X/Q) factors are used for the radiological assessment. These are presented in USAR Table 2.3-24.
- All other assumptions and input parameters are provided in Table 1 (see page 10).

The radiological results of the event are presented in Tables 2 and 3 (these results are also identical to those presented in the March 16, 1990 letter). The radiological impact of this limiting fuel drop without mitigating functions remains within the acceptable dose limitations with the longer assumed decay period. Based on these results, the concept of recently irradiated fuel is being incorporated into the APPLICABILITIES and ACTION requirements for the LCOs identified above in the "Description of Proposed Changes".

The reanalysis summarized above will be added to the PNPP USAR. This addition to the PNPP design basis will ensure that this event is considered in future reload analyses.

In addition to the offsite dose analysis that was performed to meet the guidance of the Standard Review Plan, control room doses were examined to confirm that they would not be limiting and would remain within the guidelines of General Design Criterion 19 [Ref. 6].

Plant Specific Justification

The use of the term "recently irradiated fuel" provides a mechanism for applying a cutoff in fission product decay to various specifications in which the concept applies. The seven day period to be discussed in the Technical Specification Bases has been shown by analysis to provide sufficient decay such that, assuming the design basis fuel handling accidents, radiological consequences are within the acceptance criteria of SRP 15.7.4 and General Design Criteria 19.

The revised Technical Specification APPLICABILITY requirements incorporate the term "recently irradiated" in order to establish the specific activities during which significant radioactive releases can be postulated. During MODE 4 or 5, these are:

1. Handling of recently irradiated fuel in the Primary Containment or Fuel Handling Building.

2. Operations with a potential for draining the reactor vessel (OPDRVs).

The revised requirements redefine the LCO APPLICABILITY for instrumentation and devices that isolate Primary Containment and the Fuel Handling Building, and those for ESF systems designed to mitigate the radiological impact of fuel handling accidents. The proposed APPLICABILITY is consistent with the fuel handling accident assumptions.

It should be noted that the above list of activities which "establish the specific activities during which significant radioactive releases can be postulated during MODE 4 or 5" does not include any CORE ALTERATIONS other than the fuel handling accident. The accidents postulated to occur during core alterations in addition to fuel handling accidents are: a control rod removal or withdrawal error during refueling (USAR 15.4.1.1); a control rod maloperation (USAR 15.4.3); and the inadvertent loading and operation of a fuel assembly in an improper location (USAR 15.4.7). As described in each of the above USAR discussions and in USAR Section 15.7 6.1.1, these events are not postulated to result in fuel cladding integrity damage. Therefore, no radioactive material is released from the fuel. Since the only accident postulated to occur during CORE ALTERATIONS that results in a radioactive release is the fuel handling accident, the proposed Technical Specification change which omits CORE ALTERATIONS from the APPLICABILITY of certain of the INTEGRITY and ESF system specifications is justified.

LCO APPLICABILITY requirements remain unaffected for other Specifications that are needed to prevent or mitigate CORE ALTERATION events other than the fuel handling accident. This includes specifications such as Shutdown Margin, Reactor Mode Switch (which provides controls on the refueling interlocks), SRM Instrumentation in Mode 5, and electrical power availability and distribution.

The LCO APPLICABILITY requirements for operations with a potential for draining the reactor vessel (OPDRVs) are unaffected by the proposed changes. Also, APPLICABILITY requirements are unaffected for decay heat removal systems during shutdown conditions, and for the specifications that require maintenance of high water levels over irradiated fuel.

Supplemental Plant Specific Shutdown Risk Justification

The containment and associated engineered safety feature systems are only required by the Technical Specifications during the specific activities postulated to result in a significant release of radioactivity (e.g., fuel handling accident, draindown). Technical Specification requirements are not based on providing requirements associated with "shutdown risk" considerations; instead, these are addressed through outage management administrative controls.

As discussed in a meeting between representatives of BWR-6 licensees and the NRC on July 20, 1995, the NRC was concerned about the relationship of this proposed change to the shutdown risk issues they are currently resolving. The focus of NRC concern was containment closure following a loss of decay heat removal event. Section 4.5 of NUMARC 91-06, "Guidelines For Industry Actions To Assess Shutdown Management" [Ref. 7] discusses the need to assure that containment closure can be achieved in response to a loss of decay heat removal. It identifies that the time to effect closure should be consistent with plant conditions (e.g., reactor coolant system inventory and decay heat load). Consistent with the industry's commitment in the letter from NUMARCs President, Mr. Byron Lee, Jr., to Mr. James M. Taylor of the NRC [Ref. 8], PNPP has outage management administrative controls in place for re-establishing containment closure, which are based on the recommendations of NUMARC 91-06 Section 4.5.

Also, in accordance with the Technical Specifications which control RPV water level over irradiated fuel, handling of irradiated fuel in the reactor vessel can only occur when the water level in the reactor cavity is at least 22 feet 9 inches above the top of the reactor pressure vessel flange. As illustrated during the July 20, 1995 meeting, when water volumes are this large, the risk from a loss of decay heat removal event is relatively low. Therefore, the proposed changes only reduce containment requirements during relatively low risk times during refueling outages. The proposed changes therefore do not significantly increase the shutdown risk.

Conclusions of the Safety Analysis

The Technical Specifications contain requirements which were developed to support fuel handling accident analyses which assume only a 24-hour period of reactor subcriticality prior to fuel movement (see LCO 3.9.4). Following reactor shutdown, decay of the short-lived fission products greatly reduces the fission product inventory present in irradiated fuel.

The proposed changes redefine the OPERABILITY requirements for selected accident mitigation features (INTEGRITY and mitigating ESF systems) such that these features are only required to be OPERABLE during the time frame early in a refueling outage when the limiting fuel handling accident analysis takes credit for them to function to meet the regulatory dose limits. The approach to justify these Technical Specification changes is to calculate the whole body and thyroid doses due to a fuel handling accident at various time intervals longer than 24 hours after subcriticality. The purpose is to establish a point where OPERABILITY of those features typically used to mitigate the consequences of a fuel handling accident is no longer required to meet the regulatory dose guidance (75 rem thyroid, 6 rem whole body), due to the reduced radionuclide inventory available for release. The analyses demonstrate that after several days of subcriticality, the regulatory dose guidance of SRP 15.7.4 is satisfied without credit for building INTEGRITY and mitigating ESF system operation.

The proposed changes also delete OPERABILITY requirements for INTEGRITY and mitigating ESF systems during CORE ALTERATIONS, since the only accident postulated to occur during CORE ALTERATIONS that results in a radioactive release is the fuel handling accident.

Containment closure controls are provided by outage management guidelines during periods when Technical Specification controls are not applied.

References

1. Perry Nuclear Power Plant Unit 1 Current and Improved Technical Specifications, updated through Amendment 72.

2. Perry Nuclear Power Plant Updated Safety Analysis Report, Updated through Rev. 7.

3. NUREG 0800, (Standard Review Plan), Section 15.7.4, "Radiological Consequences of Fuel Handling Accidents," Revision 1, July 1981.

4. Regulatory Guide 1.25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors", 3/23/72.

5. Letter from A. Kaplan, Vice President, Nuclear Group (CEI) to the U. S. Nuclear Regulatory Commission, dated March 16, 1990 (PY-CEI/NRR-1140L), "Technical Specification Change Request, Performance of Containment Isolation Valve Testing With Primary Containment Integrity - Shutdown".

6. 10CFR50, Appendix A, General Design Criteria 19.

7. NUMARC 91-06, Guidelines For Industry Actions To Assess Shutdown Management, December 1991.

8. Letter from Mr. Byron Lee, Jr., President and Chief Executive Officer NUMARC, to Mr. James M. Taylor, Executive Director for Operations U.S. NRC, dated December 6, 1991.

ENVIRONMENTAL CONSIDERATION

The proposed Technical Specification change request has been reviewed against the criteria of 10 CFR 51.22 for environmental considerations. As shown above and in Attachment 3, the proposed change does not involve a significant hazards consideration, does not increase the types and amounts of effluents that may be released offsite, and does not significantly increase individual or cumulative occupational radiation exposures. Based on the foregoing, it has been concluded that the proposed Technical Specification change request meets the criteria given in 10 CFR 51.22(c)(9) for a categorical exclusion from the requirement for an Environmental Impact Statement.

TABLE 1
FUEL HANDLING ACCIDENT INSIDE CONTAINMENT
PARAMETERS TABULATED FOR POSTULATED ACCIDENT ANALYSIS

	Design Basis Assumptions	Realistic Assumptions
I. Data and assumptions used to estimate radioactive sources from postulated accident.		
A. Power level	3758 MWt	Same
B. Burnup	1,000 days	Same
C. Radial peaking factor	1.5	Same
D. Fuel damage	124 rods	Same
E. Release of activity by nuclide	10% iodine 30% Kr-85 10% other noble gases	0.32% iodine 1.80% Kr-85 1.80% other noble gases
F. Radionuclide decay time	168 hours (7 days)	Same
G. Iodine gap activity species		
1. Organic	0.25%	Same
2. Inorganic	99.75%	Same
H. Minimum water coverage above damaged fuel rods	23 ft.	Same
I. Pool decontamination factors:		
1. Organic iodine	1	Same
2. Inorganic iodine	133	Same
3. Noble gases	1	Same
J. Activity Airborne in Containment	See Table 2	See Table 2
II. Data and assumptions used to estimate activity released		
A. Release pathway	Instantaneous unfiltered exhaust direct to the environment	Same
B. All other pertinent data and assumptions	Reg. Guide 1.25	Same
III. Dispersion Data (USAR Table 2.3-24)		
A. Exclusion Area Boundary (863 meters)	$4.3x10^{-4}$ sec/m^3	Same
B. Low Population Zone (4002 meters)	$4.8x10^{-5}$ sec/m^3	Same
IV. Dose Data		
A. Method of dose calculation	Reg. Guide 1.25	Same
B. Dose conversion assumptions	Reg. Guide 1.25	Same
C. Doses	See Table 3	See Table 3

TABLE 2

FUEL HANDLING ACCIDENT INSIDE CONTAINMENT

ACTIVITY AIRBORNE IN THE CONTAINMENT BUILDING AND RELEASED TO THE ENVIRONMENT (CURIES)

Isotope	Activity	
	Design	Realistic
I-131	2.05E+2	1.68E+1
I-132	*	*
I-133	1.89E+0	4.95E-2
I-134	*	*
I-135	1.90E-5	2.82E-7
Kr-83M	*	*
Kr-85M	7.54E-8	4.16E-9
Kr-85	8.95E+2	6.00E+2
Kr-87	*	*
Kr-88	*	*
Kr-89	*	*
Xe-131M	1.82E+2	4.67E+1
Xe-133M	1.35E+3	1.35E+2
Xe-33	2.52E+4	4.20E+3
Xe-135M	*	*
Xe-135	2.69E-1	5.59E-2
Xe-137	*	*
Xe-138	*	*

* Indicates isotope activity is less than E-10 curies (i.e., dose contribution is insignificant when compared to the other isotopes).

TABLE 3

FUEL HANDLING ACCIDENT INSIDE CONTAINMENT

RADIOLOGICAL EFFECTS

	Design Basis Values		Realistic Values	
	Whole Body Dose (rem)	Inhalation Dose (rem)	Whole Body Dose (rem)	Inhalation Dose (rem)*
Exclusion Area (863 meters)	1.14E-1	4.56 E+1	1.81E-2	3.73E+0
Low Population Zone (4,002 meters)	1.27E-2	5.09E+0	2.03E-3	4.16E-1

* These values are over-estimated by a factor of from 10 to 10^6 since an iodine partition factor of 100 was utilized for conservatism rather than a value of from 10^3 to 10^8 as have been experimentally determined.

SIGNIFICANT HAZARDS CONSIDERATION

This proposed amendment to the Perry Nuclear Power Plant Technical Specifications revises those specifications associated with handling irradiated fuel in Primary Containment and the Fuel Handling Building, and selected specifications associated with CORE ALTERATIONS. Specifically, analysis identifies that only "recently" irradiated fuel contains sufficient fission products to require OPERABILITY of accident mitigation features to meet the accident analysis assumptions. Analyses also show that accident mitigation features such as building INTEGRITY and engineered safety feature (ESF) ventilation systems are not required for CORE ALTERATION events. This proposed change therefore revises OPERABILITY requirements and ACTIONS for the associated Limiting Conditions for Operation. The proposed changes do not impact Technical Specification requirements for systems needed to prevent or mitigate CORE ALTERATION events other than the fuel handling accident. They also do not change requirements for systems needed to mitigate potential vessel draindown events, systems needed for decay heat removal, or the requirements to maintain high water levels over irradiated fuel.

The standards used to arrive at a determination that a request for amendment involves no significant hazards considerations are included in the Commissions regulations, 10CFR50.92. This regulation states that a proposed amendment involves no significant hazards considerations 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; (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 proposed change has been reviewed with respect to these three factors and it has been determined that the proposed change does not involve a significant hazard because:

1. *The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.*

   The proposed requirements are imposed during specific activities which can be postulated to result in significant radioactive releases. The proposed APPLICABILITY requirements are consistent with either the original design basis analyses or with revised analyses performed to support this proposed amendment. Because the equipment controlled by the revised Specifications is not considered an initiator to any previously analyzed accident, inoperability of the equipment cannot increase the probability of any previously evaluated accident.

   Consistent with the original design basis analysis, the reanalysis concludes that radiological consequences of the fuel handling accident are well within the 10 CFR 100.11 limits, as defined by acceptance criteria in Standard Review Plan Section 15.7.4. The reanalysis has previously been submitted to the Nuclear Regulatory Commission for review, and NRC confirmatory calculations reached consistent results (reference NRC Safety Evaluation for License Amendment No. 35). The results of the CORE ALTERATION events other than the fuel handling accident remain unchanged from the original design basis, which showed that these events do not result in fuel cladding integrity damage or radioactive releases. Therefore, the proposed changes do not significantly increase the consequences of any previously evaluated accident.

Based on the above, the proposed changes do not significantly increase the probability or consequences of any accident previously evaluated.

2. *The proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.*

   The proposed requirements are imposed when specific activities represent situations where significant radioactive releases can be postulated. The proposed APPLICABILITY requirements are consistent with design basis analyses. The proposed changes do not introduce any new modes of plant operation and do not involve physical modifications to the plant. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

3. *The proposed change does not involve a significant reduction in a margin of safety.*

   The proposed change imposes controls to ensure that during performance of activities which represent situations where radioactive releases are postulated, the radiological consequences are at or below the established licensing limit. Safety margins and analytical conservatisms have been evaluated and are well understood. Substantial conservatism is retained to ensure that the analysis adequately bounds all postulated event scenarios. The current margin of safety is retained.

   Specifically, the margin of safety for the fuel handling accident is the difference between the 10CFR100 limits and the licensing limit defined by the Standard Review Plan (NUREG 0800), Section 15.7.4. The licensing limit is defined by the Standard Review Plan as being "well within" the 10CFR100 limits, with "well within" defined as 25% of the 10CFR100 limits for the fuel handling accident. Excess margin is the difference between the postulated doses and the corresponding licensing limit. In the NRCs initial licensing review of the Perry Nuclear Power Plant (NUREG-0887, Section 15.3.3), the NRC accepted the design and analyses based on the results of the analyses being well within the guideline values of 10CFR100.

   The proposed APPLICABILITY requirements continue to ensure that the whole-body and thyroid doses at the exclusion area and low population zone boundaries as well as control room doses are at or below the corresponding licensing limit. The margin of safety is unchanged; therefore, the proposed changes do not involve a significant reduction in a margin of safety.

   The margin of safety for the CORE ALTERATION events other than the fuel handling accident discussed above also remains the same as in the original design basis analyses, since the proposed changes do not impact on the Technical Specification requirements for systems needed to prevent or mitigate such CORE ALTERATION events.