WOLF CREEK
NUCLEAR OPERATING CORPORATION

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ET 95-0100

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Mail Station P1-137
Washington, D. C. 20555

Subject: Docket No. 50-482: Revision to Technical Specification 3.9.4, "Containment Building Penetrations"

Gentlemen:

This letter transmits an application for amendment to Facility Operating License No. NPF-42 for Wolf Creek Generating Station (WCGS). This license amendment request proposes revising Technical Specification 3.9.4, "Containment Building Penetrations," and its associated Bases section to allow the containment personnel airlock doors to be open during core alterations and movement of irradiated fuel in containment. Also, Technical Specification Surveillance Requirement 4.9.4 would be revised to specify that each containment penetration should be in its "required condition," instead of "closed/isolated condition."

Attachment I provides a description of the proposed change along with a Safety Evaluation. Attachment II provides a No Significant Hazards Consideration Determination. Attachment III provides the Environmental Impact Determination. The specific changes to the technical specifications proposed by this request are provided as Attachment IV.

In accordance with 10 CFR 50.91, a copy of this application, with attachments, is being provided to the designated Kansas State official. This proposed revision to the WCGS Technical Specifications will be fully implemented within 30 days of formal Nuclear Regulatory Commission approval.

This license amendment request is being submitted in parallel with three other similar license amendment requests from Pacific Gas and Electric Co., Union Electric Co., and Texas Utilities Electric. The license amendment requests are being submitted in parallel to allow the NRC to review all four changes together and thereby reducing the amount of NRC resources required.

If you have any questions concerning this matter, please contact me at (316) 364-8831, extension 4553, or Mr. Richard D. Flannigan, at extension 4500.

Very truly yours,

Robert C. Hagan

Robert C. Hagan

RCH/jra

Attachments: I - Safety Evaluation
II - No Significant Hazards Consideration Determination
III - Environmental Impact Determination
IV - Proposed Technical Specification Changes

cc: G. W. Allen (KDHE), w/a
L. J. Callan (NRC), w/a
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J. C. Stone (NRC), w/a

STATE OF KANSAS)
) SS
COUNTY OF COFFEY)

Robert C. Hagan, of lawful age, being first duly sworn upon oath says that he is Vice President Engineering of Wolf Creek Nuclear Operating Corporation; that he has read the foregoing document and knows the content thereof; that he has executed that same for and on behalf of said Corporation with full power and authority to do so; and that the facts therein stated are true and correct to the best of his knowledge, information and belief.

ANGELA E. WESSEL
Notary Public - State of Kansas
My Appt. Expires 07-03-99

By Robert C. Hagan

Robert C. Hagan
Vice President
Engineering

SUBSCRIBED and sworn to before me this 22nd day of Nov. , 1995.

Angela E. Wessel

Notary Public

Expiration Date July 3, 1999

ATTACHMENT I

SAFETY EVALUATION

Safety Evaluation

Proposed Change

This license amendment request proposes revising Technical Specification 3.9.4, "Containment Building Penetrations," and its associated Bases section to allow the containment personnel airlock doors to be open during core alterations and movement of irradiated fuel in containment provided that a minimum of one door in the emergency airlock is closed and one door in the personnel airlock is capable of being closed. Also, Technical Specification Surveillance Requirement 4.9.4 would be revised to specify that each containment penetration should be in its "required condition," instead of "closed/isolated condition." This change is necessary to allow the personnel airlock doors to be open.

Background

Technical Specification 3.9.4, "Containment Building Penetrations," requires that a minimum of one containment personnel airlock door, as well as other containment penetrations, be closed during core alterations and movement of irradiated fuel assemblies within the containment. This requirement serves to contain fission product radioactivity that may be released from damaged fuel rods following a fuel handling accident, such that offsite radiation exposures are maintained well within the requirements of 10 CFR 100. This requirement is reflected in the assumptions used in the analysis (presented in Section 15.7.4 of the Updated Safety Analysis Report) that evaluates the potential radiological consequences of a fuel handling accident occurring inside containment. Since the containment shutdown purge subsystem is normally operating during refueling operations, the fuel handling accident analysis assumes that the radioactive materials from the damaged fuel rods is released to the environment via the containment shutdown purge line until it is closed. It is assumed that isolation of the containment purge line does not occur until 25 seconds after the event. After the containment is isolated, no more offsite release would occur, so the major portion of the activity release would be confined to containment. The fuel handling accident analysis also accounts for the requirements of the minimum decay time of 100 hours prior to core alterations and the minimum refueling pool water level of 23 feet over the top of the reactor vessel flange specified in Technical Specifications 3.9.3 and 3.9.10.1, respectively. These requirements ensure that the release of fission product radioactivity, subsequent to a fuel handling accident, results in doses that are well within the guideline values specified in 10 CFR 100.

During a refueling outage, other work inside containment does not stop during fuel movement and core alternations. This requires that personnel operate the containment personnel airlock doors frequently to enter and exit containment. Such heavy use of the containment personnel airlock was not anticipated during its design. As a result of this unexpectedly heavy use, failures of the door hinge pin, the door seals, the three-way equalizing valves, and other components have occurred throughout the industry. Potential failures of the containment personnel airlock doors could raise the concern that the containment personnel airlock might not be able to be sealed in the event of an accident.

The containment personnel airlock is a welded steel assembly with two doors with double gaskets in series. The containment personnel airlock barrel is inserted through the existing containment wall sleeves; then the attachment collars furnished with the airlock is welded to the sleeves. The containment personnel airlock barrel has a 9 foot 11 inches inside diameter with sufficient length to provide a minimum clear distance of approximately 8 feet between the doors.

The two doors for the containment personnel airlock are electrically and mechanically interlocked so that one door cannot be opened unless the second door is sealed. A pressure-equalizing valve at each door is provided to equalize pressure across the doors when personnel are entering or leaving the containment. The valves are properly interlocked so that they both cannot be open at the same time and each valve can be operated only when the opposite door is closed and locked. Provisions are made to bypass the interlock to permit both doors to be opened when safe to do so.

From a practical standpoint, Technical Specification 3.9.4 will not prevent all radioactive releases from the containment following a postulated fuel handling accident. There are a large number of people in the containment during a refueling outage, even during fuel movement and core alterations. Should a fuel handling accident occur, it would take a number of cycles of the containment personnel airlock to evacuate personnel from within containment. With each containment personnel airlock cycle, more containment air would be released. While waiting for their turn to exit, the workers would be exposed to the released activity. Alternatively, the Shift Supervisor could invoke 10 CFR 50.54(x), order both doors of the containment personnel airlock opened while the personnel in the containment are evacuated, and then close the doors. In either case, there would be a release of activity into the atmosphere. Under the proposed change, the containment could be evacuated without invoking 10 CFR 50.54(x) and then sealed. This would reduce the dose to workers in the event of an accident while maintaining acceptable doses to the public.

If the containment personnel airlock doors are allowed to be open during core alterations or fuel movement, the above concerns could be resolved. However, this proposed technical specification amendment could potentially result in an increased dose consequence when both doors of the containment personnel airlock are open at the time of the postulated accident. To justify the proposed change, the fuel handling accident inside containment was reanalyzed to confirm that the potential doses to the public and the control room operators would remain within acceptable limits should a fuel handling accident occur when the personnel airlock doors are left open during core alterations or movement of irradiated fuel.

Evaluation

During core alterations or movement of irradiated fuel within containment, the most severe radiological consequences result from a fuel handling accident. The fuel handling accident is a postulated event that involves damage to irradiated fuel. Fuel handling accidents include the dropping of a single irradiated fuel assembly and handling tool or a heavy object onto other irradiated fuel assemblies.

To assess the potential radiological consequences resulting from the occurrence of a postulated fuel handling accident when the personnel airlock doors are left open during core alterations or fuel movement, a revised dose calculation was performed. Specifically, the total-body dose due to immersion from direct radiation and the thyroid dose due to inhalation was calculated for the 0-2 hour time period at the exclusion area boundary and for the duration of the accident at the low-population zone outer boundary.

The assumptions postulated in the revised calculation of the radiological consequences of a fuel handling accident inside containment during core alterations or fuel movement are consistent with the assumptions of 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," and are summarized below:

a. The accident is postulated to occur 100 hours after shutdown, which is the earliest time fuel handling operations may begin (Technical Specification 3.9.3). Radioactive decay of the fission product inventory during this time period is taken into consideration.

b. The minimum water depth between the top of the damaged fuel rods and the refueling pool surface is 23 feet.

c. The dropped fuel assembly is assumed to be the assembly containing the peak fission product inventory. All the fuel rods contained in the dropped assembly are assumed to be damaged. In addition, the dropped assembly is assumed to damage 20 percent of the rods of an additional assembly.

d. The values assumed for individual fission product inventories in the damaged assembly were calculated based upon a radial peaking factor of 1.65.

e. Damaged rods are assumed to release their gap activities. The gap activity released to the refueling pool from the damaged fuel rods consists of 10 percent of the total noble gases other than Kr-85, 30 percent of the Kr-85, and 10 percent of the total radioactive iodine contained in the fuel rods at the time of the accident.

f. The retention of noble gases in the pool is negligible (i.e., decontamination factor of 1).

g. The effective refueling pool decontamination factor for iodine is taken as 100 (i.e., 99 percent of the total iodine released from the damaged rods is retained by the refueling pool water).

h. The gaseous effluent escaping from the refueling pool in containment is assumed to be released to the environment through the open personnel airlock and the adjacent auxiliary building without mixing in the surrounding atmosphere.

i. The auxiliary building atmosphere is normally exhausted through filter absorbers designed to remove iodine. However, no credit it taken for iodine removal by the atmosphere filtration system filters.

j. The radioactive material that escapes from the refueling pool to the reactor containment building is released from the building over a two-hour time period.

Using these assumptions and parameters, the potential radiological consequences resulting from a postulated fuel handling accident were recalculated and the results are presented in Table 1. The potential radiological consequences as a result of a fuel handling accident coincident with the containment personnel airlock doors remaining open are higher than that of the current licensing basis analysis. However, the potential doses are still well within the guideline values of 10 CFR 100 for the whole-body and thyroid doses. The potential radiation doses to control room personnel for the postulated fuel handling accident were also calculated. The resultant thyroid dose to control room personnel was calculated to be 9.7 rem and is within the exposure guidelines of General Design Criterion 19.

Table 1 Radiological Consequences of A Postulated Fuel Handling Accident with the Containment Personnel Airlock Doors Remaining Open during Core Alterations or Fuel Movement

	Dose (rem)	
	Revised	**Current USAR**
Site Boundary (0-2 hr):		
Thyroid	5.52E+1	1.62E+1
Whole Body	1.86E-1	5.5E-2
Low-Population Zone (Duration):		
Thyroid	7.36E+0	2.2E+0
Whole body	2.47E-2	7.3E-3

Precedents

Similar license amendments have been approved or have been submitted and are awaiting approval. In particular, Baltimore Gas and Electric had a similar change approved for Calvert Cliffs Nuclear Power Plant. The significant differences between Calvert Cliffs change and the change proposed herein is that the Calvert Cliffs Technical Specifications require that: 1) an individual be designated to close the operable airlock door in the event of a fuel handling accident, 2) the plant be in Mode 6, and 3) there is 23 feet of water above the fuel.

The requirement to have an individual designated to close the personnel airlock is not included in this proposed change. The reason for the difference is that the stationing of an individual to close the airlock door at Calvert Cliffs was considered a conservative measure to deal with the plant specific design feature that the airlock does not open into an area whose exhaust is filtered. At Wolf Creek Generating Station, the airlock opens into an area of the auxiliary building which is exhausted through filters in the ventilation system.

The requirement to have the plant in Mode 6 is not included in this proposed change. The requirement is redundant since Technical Specification 3.9.4 is applicable only during core alterations and movement of irradiated fuel. As a result, the plant by definition must be in Mode 6.

The requirement to maintain 23 feet of water above the fuel was not included in this proposed change. The requirement would be redundant since Technical Specification 3.9.10.1 places restrictions on the required minimum refueling pool water level during movement of irradiated fuel within containment. Also, Technical Specification 3.9.10.2, which is being relocated to the Operational Requirements Manual (Chapter 16 of the USAR) in accordance with Amendment No. 89, places restrictions on the required minimum refueling pool water level during movement of control rods within the reactor pressure vessel.

Conclusions

The proposed change represents the potential for increased offsite doses because the containment personnel airlock doors are assumed to be open at the time of the accident. However, the results of the dose re-analysis indicate that the potential dose consequences would remain below the acceptable regulatory limits even if the accident should occur coincident with the personnel airlock doors being open during core alterations or movement of irradiated fuel assemblies. The increase in doses would be offset by the decreased potential radiation dose to workers in the event of a fuel handling accident, and the increased reliability of the containment personnel airlock door in the event of an accident.

While allowing both containment personnel airlock doors to remain open during core alterations or movement of irradiated fuel within containment, the proposed change contains requirements that assure a minimum of one door in the emergency airlock is closed and one door in the personnel airlock is capable of being closed. The above requirements would serve the purpose of minimizing the release of radioactive material and the resultant offsite dose consequences.

Based on the above discussions and the considerations presented in Attachment II, the proposed change does not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report; or create a possibility for an accident or malfunction of a different type that any previously evaluated in the safety analysis report; or reduce the margin of safety as defined in the basis for any technical specification. Therefore, the proposed change does not adversely affect or endanger the health or safety of the general public or involve a significant safety hazard.

ATTACHMENT II

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

No Significant Hazards Consideration Determination

This license amendment request proposes revising Technical Specification 3.9.4, "Containment Building Penetrations," and its associated Bases section to allow the containment personnel airlock doors to be open during core alterations and movement of irradiated fuel in containment provided that a minimum of one door in the emergency airlock is closed and one door in the personnel airlock is capable of being closed. Also, Technical Specification Surveillance Requirement 4.9.4 would be revised to specify that each containment penetration should be in its "required condition," instead of "closed/isolated condition."

Standard I - Involves a Significant Increase in the Probability or Consequences of an Accident Previously Evaluated

The proposed change to Technical Specification 3.9.4 would allow the containment personnel airlock to be open during fuel movement and core alterations. The containment personnel airlock is currently closed during fuel movement and core alterations to prevent the escape of radioactive material in the event of a fuel handling accident. The containment personnel airlock is not an initiator of any accident. Whether the containment personnel airlock doors are open or closed during fuel movement and core alterations has no affect on the probability of any accident previously evaluated.

The proposed change does alter assumptions previously made in evaluating the radiological consequences of the fuel handling accident inside the containment building. The proposed change allows for the containment personnel airlock to be open during refueling. The radiological consequences described in this change are bounded by those given in the Wolf Creek Generating Station Safety Evaluation Report and General Design Criteria 19. All doses for the proposed change are less than the acceptance criteria, therefore, there is no significant increase in the consequences of an accident previously analyzed.

The proposed change would significantly reduce the dose to workers in the containment in the event of a fuel handling accident by accelerating the containment evacuation process. The proposed change would also significantly decrease the wear on the containment personnel airlock doors and, consequently, increase the reliability of the containment personnel airlock doors in the event of an accident.

Since the probability of a fuel handling accident is unaffected by the airlock door positions, and the increased doses do not exceed acceptance limits, operation of the facility in accordance with the proposed amendment would not affect the probability or consequences of an accident previously analyzed.

Standard II - Create the Possibility of a New or Different Kind of Accident from any Previously Evaluated

The proposed change affects a previously evaluated accident, e.g., a fuel handling accident inside containment. The existing accident has been modified to account for the containment personnel airlock doors being opened at the time of the accident. It does not represent a significant change in the configuration or operation of the plant. Therefore, operation of the facility in accordance with the proposed amendment would not create the possibility of a new or different kind of accident from any accident previously evaluated.

Standard III - Involve a Significant Reduction in the Margin of Safety

The margin of safety is reduced when the offsite and control room doses exceed the acceptance criteria in the Wolf Creek Generating Station Safety Evaluation Report. As previously discussed in the response to Standard I, the offsite and control room doses are below the acceptance criteria. Therefore, operation of the facility in accordance with the proposed amendment would not involve a significant reduction in the margin of safety.

ATTACHMENT III

ENVIRONMENTAL IMPACT DETERMINATION

Environmental Impact Determination

10 CFR 51.22(b) specifies the criteria for categorical exclusions from the requirements for a specific environmental assessment per 10 CFR 51.21. This amendment request meets the criteria specified in 10 CFR 51.22(c)(9). The specific criteria contained in this section are discussed below.

(i) the amendment involves no significant hazards consideration

As demonstrated in the No Significant Hazards Consideration Determination in Attachment II, the requested license amendment does not involve any significant hazards consideration.

(ii) there is no significant change in the types or significant increase in the amounts of any effluents that may be released offsite

The requested license amendment involves no change to the facility and does not involve any change in the manner of operation of any plant systems involving the generation, collection or processing of radioactive materials or other types of effluents. Therefore, no increase in the amounts of effluents or new types of effluents would be created.

(iii) there is no significant increase in individual or cumulative occupational radiation exposure

The requested license amendment involves no change to the facility and does not involve any change in the manner of operation of any plant systems involving the generation, collection or processing of radioactive materials or other types of effluents. Furthermore, implementation of this proposed change will not involve work activities which could contribute to occupational radiation exposure. Therefore, there will be no increase in individual or cumulative occupational radiation exposure associated with this proposed change.

Based on the above it is concluded that there will be no impact on the environment resulting from this change. The change meets the criteria specified in 10 CFR 51.22 for a categorical exclusion from the requirements of 10 CFR 51.21 relative to specific environmental assessment by the Commission.

ATTACHMENT IV

PROPOSED TECHNICAL SPECIFICATION CHANGES