REVISED RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

APR1400 Design Certification

Korea Electric Power Corporation / Korea Hydro & Nuclear Power Co., LTD

Docket No. 52-046

RAI No.:	**481-8546**
SRP Section:	**16 – Technical Specifications**
Application Section:	**16.3.4, 16.3.5, 16.3.6, 16.3.7, 16.3.9**
Date of RAI Issue:	**05/12/2016**

Question No. 16-150

1. The staff considered the response to Subquestion 1 incomplete for the following reason.

   In the original RAI, the staff raised the Subquestion 1 issue as follows (with underlined text for emphasis):

   "APR1400 TS Table 3.7.1-2 contains a Note that allows the "as-left" tolerance for the lift setting to be +3%. In the NUREG-1432, this allowance is discussed in the TS Bases for SR 3.7.1.1, with the +3% placed in brackets indicating further supporting information to meet ASME Code, Section III, NC 7000 requirements. ASME Code, Section III, NC 7512.2, “Set Pressure Tolerance,” states, in part, “ … The set pressure tolerance plus or minus shall not exceed the following: 2 psi (15 kPa) for pressures up to and including 70 psi (480 kPa), 3% for pressures from 70 psi (480 kPa) to 300 psi (2 MPa), 10 psi (70 kPa) for pressures over 300 psi (2 MPa) to 1,000 psi (7 MPa), and 1% for pressures over 1,000 psi (7 MPa). The set pressure tolerance shall apply unless a greater tolerance is established as permissible in the Overpressure Protection Report (NC-7200) and in the safety valve Design Specification (NCA-3250).” The applicant is requested to provide a reference to the applicable documents in the TS Bases."

   It should be noted that in the above quotation, the staff made an error when stating the +3% tolerance is applicable to an “as-left” value. In NUREG-1432, for an “as-found” value the tolerance is +3%, and for an “as-left” value the tolerance is +1%. As a result, in the response, the applicant provided the following information:

   The “as-left” tolerance for the Main Steam Safety Valve lift setting of ±3% stated in TS Table 3.7.1-2 is described in “I-1350 Test Frequency, Classes 2 and 3 Pressure Relief Valves” of the ASME OM Code (Reference 4) and in Part 1, Section 1.3.4.1 “Pressure Relief Valves” of ANSI/ASME OM-1-1987 (Reference 5) which are referenced in the Bases section for SR 3.7.1.1.

The applicant is requested to address the original stated issue given the required ASME

Code +1% tolerance is for MSSV lift setting “as-found” values.

2. The staff found the response to Subquestion 3 acceptable, however, the applicant is requested to correct editorial errors in the proposed change to Subsection B 3.7.5 Background section as indicated in the following markup:

   The **two** auxiliary feedwater (AFW) pumps in each mechanical division take suction from a respective **common** auxiliary feedwater storage tank (AFWST) ~~and have a respective discharge header~~ (LCO 3.7.6)**, each pump with a respective discharge header,** and ~~pump~~ **discharge** to a respective steam generator secondary side through a common AFW **discharge** header, which ~~connects~~ to the steam generator downcomer main feedwater (MFW) piping inside containment.

3. The staff considered the response to Subquestion 7 incomplete in that, for clarification of the need to have all four 100% AFW pumps to be OPERABLE, Subsection B 3.7.5 LCO section should be revised to include a discussion of the following accident scenario, which was presented in the response, with suggested changes for clarity indicated by the markup:

   Assuming a postulated pipe failure concurrent with a single active component failure, four 100 percent capacity pumps are required to be OPERABLE for the AFW system. If one steam generator is not OPERABLE for reactor cooling on an initiating event, the turbine driven pump and the motor-driven pump in that mechanical division are also not OPERABLE due to the respective inoperable steam generator. Concurrent with the initiating event, a single active component failure is considered for the turbine-driven pump or the motor driven pump in the other mechanical division. One AFW pump and the associated SG would remain OPERABLE to provide reactor cooling because of the AFW system design that provides redundant capacity, and motive power that is both independent and diverse. The two 100 percent capacity motor-driven pumps are powered from independent emergency buses and each of the two 100 percent capacity turbine-driven pumps are powered from steam supplied by the respective SG, which provides diversity. This is accomplished by powering two 100 percent capacity motor-driven pumps from independent emergency buses and by a diverse means of steam supply for the two 100 percent capacity turbine-driven pumps.

   The applicant is requested to include the above clarifying details in the TS Bases.

4. The staff considered the response to Subquestion 14 incomplete for the following reason.

   In the original RAI, the staff raised the Item 14 issue as follows:

   > LCO 3.7.16 states “The combination of initial enrichment and burnup of each spent fuel assembly stored in Region II shall be within the acceptable burnup domain of Figure 3.7.16-1 or in accordance with Specification 4.3.1.1.

> The APR1400 TS 3.7.16 provisions and their associated supporting information in the TS Bases are not consistent with guidance in the STS in that complete information regarding NRC-approved documents for the high-density (Region II) storage of the spent fuel assemblies is not provided in either the TS Bases or TS 4.3 provisions. The applicant is requested to address this missing information.

In the response, the applicant provided the following information:

> Only the Region I spent fuel storage racks are designed to store fuel with a discharge burnup in the "unacceptable domain" of Figure 3.7.16-1. Since TS 3.7.16 and its associated Bases reference Specification 4.3.1.1, TS 4.3.1.1.f will be revised to limit new or partially spent fuel assemblies with discharge burnup in the "unacceptable domain" of Figure 3.7.16-1 to only be stored in Region I spent fuel racks.

The staff reviewed the proposed change to TS 4.3.1.1.f, that is presented as follows:

> New or partially spent fuel assemblies with a discharge burnup in the "unacceptable domain" of Figure 3.7.16-1 New or partially spent fuel assemblies with a discharge burnup in the "unacceptable domain" of Figure 3.7.16-~~1 will be stored in compliance with the NRC-approved specific~~ document ~~containing the analytical methods, title, date, or specific configuration or figure~~ shall only be stored in Region I spent fuel storage racks.

This change appears to negate an earlier proposed change in the response to RAI 93-8075, Question 16-1 that was presented as follows:

> New or partially spent fuel assemblies with a discharge burnup in the "unacceptable domain" of Figure 3.7.16-1 will be stored New or partially spent fuel assemblies with a discharge burnup in the "unacceptable domain" of Figure 3.7.16-1 will be stored only in the Region I of spent fuel storage rack(s) in compliance with ~~the NRC approved specific document containing the analytical methods, title, date, or specific configuration or figure~~ the technical report titled "Criticality Analysis of New and Spent Fuel Storage Racks".

The applicant is requested to resolve the above discrepancy.

In the response to RAI 93-8075, Question 16-1, the applicant also proposed the following change to TS 4.3.1.1.b:

> K_{eff} < 1.0 if ~~fully~~ flooded with unborated water and K_{eff} ≤ 0.95 if flooded with borated water at a minimum soluble boron concentration described in the LCO 3.7.15, which includes an allowance for uncertainties ~~as described in Section 9.1, "Fuel Storage and Handling."~~;

The staff understands the minimum boron concentration that is credited in the criticality analysis is addressed in Technical Report (TeR) APR1400-Z-A-NR-14001-P, Rev.0,

and is applicable only to fuel assemblies stored in Region II storage racks (e.g., mis-loading of an unirradiated fuel assembly or an insufficient depleted fuel assembly). Therefore, the applicant is requested to consider the following recommendations:

A. Revise the proposed change to TS 4.3.1.1.b to reflect the design criteria for spent fuel storage racks described in DCD Subsection 9.1.1.

B. Add a Figure to Section 4.3 showing the physical lay-out of the spent fuel pool that clearly identifies the locations of Region I and Region II storage racks within the spent fuel pool.

C. Revise the Background section of the Bases for TS 3.7.16 to include a discussion of the minimum boron concentration credited in the criticality analysis that is described in DCD Subsection 9.1.1.

D. Add TeR APR1400-Z-A-NR-14001-P, Rev.0 to the References section of the Bases for TS 3.7.16.

Response – (Rev. 2)

1. In the APR1400, the MSSV lift setting “as-left” values is ±1% in accordance with NC-7000, ASME Sec. III. These values are tested and verified at the valve supplier’s shop.

   After the completion of plant construction, these values of ±1% on Table 3.7.1-2 are verified in accordance with inservice testing program based on SR 3.7.1.1 described on DCD TS 3.7.1. Following testing, lift setting will be within ±1%.

   In regards to the Note on Table 3.7.1-2 in TS 3.7.1 ;

   Table 3.7.1-2 allows a ±3% setpoint tolerance for OPERABILITY. Therefore further testing for the valves is unnecessary in the case where the lift setting is within OPERABILITY limit. If the lift setting does not meet the OPERABILITY limit, two additional valves (maximum of the total number of MSSV) per valve with unsatisfactory result are required to be tested according to the ANSI/ASME OM-1987 requirements.

   Generic TS 3.7.1, Table 3.7.1-2 and, Base 3.7.1 will be revised as indicated in the attachment.

2. Editorial clarification in the response to RAI 120-7977, Subquestion 3 will be revised.

3. TS Bases B 3.7.5 LCO section will be revised to include suggested changes for clarify.

4-A. TS 4.3.1.1.b, e and f will be revised as indicated in the attached markup.

4-B. Figure 4.3-1 will be added to identify the locations of Region I and Region II storage racks within the spent fuel pool.

4-C. Background section of the Bases for TS 3.7.16 will be revised to include a discussion of the minimum soluble boron concentration credited in the criticality analysis.

4-D. Criticality analysis TeR (APR1400-Z-A-NR-14001-P) will be added to the reference section of the Bases for TS 3.7.16.

Impact on DCD

Same as changes described in the Impact on Technical Specifications section.

Impact on PRA

There is no impact on the PRA.

Impact on Technical Specifications

TS 3.7.1, Table 3.7.1-2 and Bases for TS 3.7.1 will be revised as indicated in the attachment.

The Bases for TS 3.7.5 will be revised as indicated in the attachment.

TS 4.3.1.1 and Bases for TS 3.7.16 will be revised as shown in the attachment.

Impact on Technical/Topical/Environmental Reports

There is no impact on any Technical, Topical, or Environmental Report.

RAI 481-8546 - Question 16-150

RAI 481-8546 - Question 16-150_Rev.1

Table 3.7.1-2 (page 1 of 1)
Main Steam Safety Valve Lift Settings

Valve Number		Lift Setting (psig ±1 %)
Steam Generator #1	Steam Generator #2	
V1301	V1311	1,174
V1303	V1313	1,205
V1305	V1315	1,230
V1307	V1317	1,230
V1309	V1319	1,230
V1302	V1312	1,174
V1304	V1314	1,205
V1306	V1316	1,230
V1308	V1318	1,230
V1310	V1320	1,230

~~NOTE~~

~~Table 3.7.1-2 allows a ± 3 % setpoint tolerance for OPERABILITY ; however, the valves are reset to ± 1 % during the Surveillance to allow for drift.~~

Replace

~~The MSSVs lift setpoint on Table 3.7.1-2 shall be within ±3% of OPERABILITY limit and the valve shall be reset within the setpoint limit of ±1% per Table 3.7.1-2 in accordance with the In-service Testing Program if the lift setting exceeds the setpoint limit of ±1%.~~

Replace

Each MSSV's as-found lift setting shall be within ±3% of the lift setting value stated in Table 3.7.1-2 for the valve to be considered OPERABLE. The valve's lift setting shall be reset to within the calibration tolerance of ±1% of the lift setting value stated in Table 3.7.1-2 if the lift setting is found to be outside the calibration tolerance.

RAI 439-8524 - Question 16-130

RAI 481-8546 - Question 16-150

RAI 481-8546 - Question 16-150_Rev.1

RAI 481-8546 - Question 16-150_Rev.2

in MODES 1 and 2.

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.1.1	------------------------ NOTE ------------------------ Only required to be performed ~~in MODE 3~~. In case of entering MODES 3 and 4 for lift setting and test of MSSV, SR 3.0.4 would not apply. Verify each required MSSV lift setpoint per Table 3.7.1-2 in accordance with Inservice Testing Program. Following testing, lift settings shall be within ± 1 %.	In accordance with Inservice Testing Program.

Delete

~~Verify each required MSSV lift setpoint is within ± 3% of the OPERABILITY limit and reset the valve within the setpoint limit per Table 3.7.1-2 in accordance with the In-service Testing Program if the lift setting exceeds the setpoint limit of ± 1%.~~

Replace

Verify each required MSSV is within ±3% of the lift setting value stated in Table 3.7.1-2, in accordance with the In-service Testing Program. If the lift setting is found to be outside the calibration tolerance of ±1% of the lift setting value stated in Table 3.7.1-2, the valve lift setting shall be reset to within the calibration tolerance.

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BASES

APPLICABLE SAFETY ANALYSES (continued)

The maximum relieving rate during the LOCV event is less than the rated capacity of two MSSVs.

The limiting accident for peak RCS pressure is the full power feedwater line break (FWLB), inside containment, with the failure of the backflow check valve in the feedwater line from the affected steam generator. Water from the affected steam generator is assumed to be lost through the break with minimal additional heat transfer from the RCS. With heat removal limited to the unaffected steam generator, the reduced heat transfer causes an increase in RCS temperature, and the resulting RCS fluid expansion causes an increase in pressure. The RCS peak pressure is less than 110 % of the design pressure of 2500 psia with the pressurizer safety valves providing relief capacity. The maximum relieving rate of the MSSVs during the FWLB event is less than the rated capacity of two MSSVs.

In safety analysis, the lift setpoint of MSSV is considered total ± 4% uncertainty that includes ± 3% setpoint uncertainty that includes ± 3% setpoint uncertainty with long term drift and ± 1% instrument error conservatively.

Using conservative analysis assumptions, a small range of FWLB sizes less than a full double ended guillotine break produce an RCS pressure exceeding 110 % (2750 psia) of design pressure. This is considered acceptable as RCS pressure is still well below 120 % of design pressure where deformation could occur. The probability of this event is in the range of 4×10^{-6}/year.

The MSSVs satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

This LCO requires all MSSVs to be OPERABLE in compliance with Reference 2, even though this is not a requirement of the DBA analysis. This is because operation with less than the full number of MSSVs requires limitations on allowable THERMAL POWER (to meet Reference 2 requirements), and adjustment to the reactor protection system trip setpoints. These limitations are according to those shown in Table 3.7.1-1, Required Action A.1, and Required Action A.2 in the accompanying LCO. An MSSV is considered inoperable if it fails to open upon demand.

The OPERABILITY of the MSSVs is defined as the ability to fully open within the setpoint tolerances, relieve steam generator overpressure, and reseat when pressure has been reduced. The OPERABILITY of the MSSVs is determined by periodic surveillance testing in accordance with the inservice testing program.

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BASES

ACTIONS (continued)

B.1 and B.2

If the MSSVs cannot be restored to OPERABLE status in the associated Completion Time, or if one or more steam generators have less than four MSSVs OPERABLE, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4 within 24 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.1.1

This SR verifies the OPERABILITY of the MSSVs by the verification of each MSSV lift setpoints in accordance with the Inservice Testing Program. The ASME Code (Reference 4), requires that safety and relief valve tests be performed in accordance with ANSI/ASME OM-1-1987 (Reference 5). According to Reference 5, the following tests are required for MSSVs:

a. Visual examination

b. Seat tightness determination

c. Setpoint pressure determination (lift setting)

d. Compliance with owner's seat tightness criteria

e. Verification of the balancing device integrity on balanced valves

The ANSI/ASME Standard requires that all valves be tested every 5 years, and a minimum of 20 % of the valves be tested every 24 months. The ASME Code specifies the activities and frequencies necessary to satisfy the requirements. Table 3.7.1-2 allows a ± 3 % setpoint tolerance for OPERABILITY; however, the valves are reset to ± 1 % during the Surveillance to allow for drift.

Table 3.7.1-2 allows a ±3% setpoint tolerance for OPERABILITY. Therefore further testing for the valves is unnecessary in the case where the lift setting is within OPERABILITY limit. If the lift setting does not meet the OPERABILITY limit, two additional valves (maximum of the total number of MSSV) per valve showing an unsatisfactory result are required to be tested according to the ANSI/ASME OM-1987 requirements. In case that the lift setting is not within ±1% even though it is within the OPERABILITY limit, the valves are reset within ±1%

RAI 120-7977 Question 16-24

~~RAI 439-8524 - Question 16-130~~

RAI 481-8546 - Question 16-150

RAI 481-8546 - Question 16-150_Rev.2

B 3.7 PLANT SYSTEMS

B 3.7.5 Auxiliary Feedwater System (AFWS)

BASES

BACKGROUND

The AFWS automatically supplies feedwater to the steam generators to remove decay heat from the reactor coolant system upon the loss of normal feedwater supply. ~~The auxiliary feedwater (AFW) pumps take suction through separate and independent suction lines from the auxiliary feedwater storage tanks (AFWSTs) (LCO 3.7.6) and pump to the steam generator secondary side via a separate and independent connection to the main feedwater (MFW) piping inside containment.~~ The steam generator functions as a heat sink for core decay heat. The heat load is dissipated by releasing steam to the atmosphere from the steam generators via the main steam safety valves (MSSVs) (LCO 3.7.1) or main steam atmospheric dump valves (MSADVs) (LCO 3.7.4). If the main condenser is available, steam may be released to the main condenser via the turbine bypass valves.

The AFWS consists of two motor driven AFW pumps and two steam turbine driven pumps configured into four trains. Each motor driven pump provides 100 % of AFW flow capacity and each turbine driven pump provides 100 % of the required capacity to its respective steam generator as assumed in the accident analysis. The pumps are equipped with independent recirculation lines to prevent pump operation against close system.

Each motor driven AFW pump is powered from an independent Class 1E power supply, and feeds one steam generator. One pump at full flow is sufficient to remove decay heat and cool the unit to shutdown cooling system (SCS) entry conditions.

Each turbine driven AFW pump receives steam from an independent main steam line, upstream of the main steam isolation valve (MSIV). Each of the steam feed lines will supply 100 % of the requirements of the turbine driven AFW pump. The turbine driven AFW pump supplies feedwater to the steam generator which provides driving steam, with DC-powered control valves actuated by the auxiliary feedwater actuation signal (AFAS).

The two auxiliary feedwater (AFW) pumps in each mechanical division take suction from a respective common auxiliary feedwater storage tank (AFWST) ~~and have a respective discharge header.~~ (LCO 3.7.6) ~~and pump~~, each with a respective discharge header, and pump discharge to a respective steam generator secondary side through a common AFW discharge header, which connects to the steam generator downcomer main feedwater (MFW) piping inside containment.

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BASES

APPLICABLE SAFETY ANALYSES (continued)

The AFWS design is such that it can perform its function following an FWLB between the main feed water isolation valve and containment, combined with a loss of offsite power following turbine trip, and a single active failure of the turbine driven AFW pump. The AFW flow to the faulted steam generator is terminated manually by the operator action. Sufficient flow would be delivered to the intact steam generator by the redundant AFW pump.

The AFWS satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The LCO requires that four independent AFW trains be OPERABLE to ensure that the AFWS will perform the design safety function to mitigate the consequences of accidents that could result in overpressurization of the reactor coolant pressure boundary. Four independent AFW pumps, in four diverse trains, ensure availability of residual heat removal capability for all events accomplished by a loss of offsite power and a single failure. This is accomplished by powering two pumps from independent emergency buses. The third and fourth AFW pumps are powered by a diverse means, two steam driven turbines supplied with steam from an independent source not isolated by the closure of the MSIVs.

The AFWS is considered to be OPERABLE when the components and flow paths required to provide AFW flow to the steam generators are OPERABLE. This requires that the two motor driven AFW pumps be OPERABLE in two diverse paths, each supplying AFW flow to a separate steam generator. Two turbine driven AFW pumps shall be OPERABLE with steam supplies from the main steam lines upstream of the MSIVs, and each capable of supplying AFW flow to the steam generators which provides driving steam. The piping, valves, instrumentation, and controls in the required flow paths shall also be OPERABLE.

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The LCO is modified by a Note indicating that one AFW train, which includes a motor driven pump, is required to be OPERABLE in MODE 4 when a steam generator is relied upon for heat removal. This is because of reduced heat removal requirements, the short period of time in MODE 4 during which AFW is required, and the insufficient steam supply available in MODE 4 to power the turbine driven AFW pump.

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Assuming a postulated pipe failure concurrent with a single active component failure, four 100 percent capacity pumps are required to be OPERABLE for the AFW system. If one steam generator is not OPERABLE for reactor cooling on an initiating event, the turbine driven pump and the motordriven pump in that mechanical division are also not OPERABLE due to the respective inoperable steam generator. Concurrent with the initiating event, a single active component failure is considered for the turbine-driven pump or the motor driven pump in the other mechanical division. One AFW pump and the associated SG would remain OPERABLE to provide reactor cooling because of the AFW system design that provides redundant capacity, and motive power that is both independent and diverse. The two 100 percent capacity motor-driven pumps are powered from independent emergency buses and each of the two 100 percent capacity turbine-driven pumps are powered from steam supplied by the respective SG, which provides diversity. This is accomplished by powering two 100 percent capacity motor-driven pumps from independent emergency buses and by a diverse means of steam supply for the two 100 percent capacity turbine-driven pumps.

RAI 93-8075 Question 16-1

RAI 481-8546 Question 16-150

4.0 DESIGN FEATURES

4.1 Site Location

[Text description of site location.]

4.2 Reactor Core

4.2.1 Fuel Assemblies

The reactor shall contain 241 fuel assemblies. Each assembly shall consist of a matrix of zirconium alloy clad fuel rods with an initial composition of natural or slightly enriched uranium dioxide (UO_2) as fuel material. Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in non-limiting core regions.

4.2.2 Control Rod Assemblies

water borated to 1231 ppm enriched, which includes an allowance for uncertainties;

The reactor core shall contain 81 full strength and 12 part strength control element assemblies (CEAs). The control material of full strength and part strength CEA shall be boron carbide and Inconel Alloy 625, respectively.

4.3 Fuel Storage

K_{eff} < 1.0 if flooded with unborated water and K_{eff} • 0.95 if flooded with ~~borated water at a minimum soluble boron concentration described in the LCO 3.7.15, which includes an allowance for uncertainties;~~

4.3.1 Criticality

4.3.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. Fuel assemblies having a maximum U-235 enrichment of 5 weight percent;

b. ~~K_{eff} < 1.0 if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.1, "Fuel Storage and Handling.";~~

c. A nominal (27.5 cm (10.83 in)) center-to-center distance between fuel assemblies placed in the Region I of spent fuel storage racks;

d. A nominal (22.5 cm (8.86 in)) center-to-center distance between fuel assemblies placed in the Region II of spent fuel storage racks

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Fuel assemblies

4.0 DESIGN FEATURES

e. ~~New or partially spent fuel assemblies~~ with a discharge burnup in the "acceptable domain" of Figure 3.7.16-1 may be allowed unrestricted storage in Region I or Region II of ~~spent fuel storage rack(s);~~ and

Figure 4.3-1

f. ~~New or partially spent fuel assemblies with a discharge burnup in the "unacceptable domain" of Figure 3.7.16-1 will be stored in compliance with the NRC-approved specific document containing the analytical methods, title, date, or specific configuration or figure.~~

4.3.1.2 The new fuel storage racks are designed and shall be maintained with:

a. Fuel assemblies having a maximum U-235 enrichment of 5 weight percent;

b. $K_{eff} \leq 0.95$ if fully flooded with unborated water, or mist, which includes an allowance for uncertainties as described in Section 9.1 “Fuel Storage and Handling.”;

c. $K_{eff} \leq 0.98$ if moderated by aqueous foam, which includes an allowance for uncertainties as described in Section 9.1 “Fuel Storage and Handling.”; and

d. A nominal center-to-center distance between fuel assemblies placed in the new fuel storage racks is 35.5 cm (14 in).

4.3.2 Drainage

The spent fuel pool is designed and shall be maintained above 7m (23 ft) from the top of the spent fuel storage rack to prevent inadvertent draining.

4.3.3 Capacity

The spent fuel pool is designed and shall be maintained with a storage capacity limited to no more than 1,792 fuel assemblies.

New or partially spent fuel assemblies with a discharge burnup in the “unacceptable domain” of Figure 3.7.16-1 ~~will be stored only in the Region I of spent fuel storage rack(s) in compliance with the technical report titled "Criticality Analysis of New and Spent Fuel Storage Racks"~~.

shall be stored only in the Region I of Figure 4.3-1.

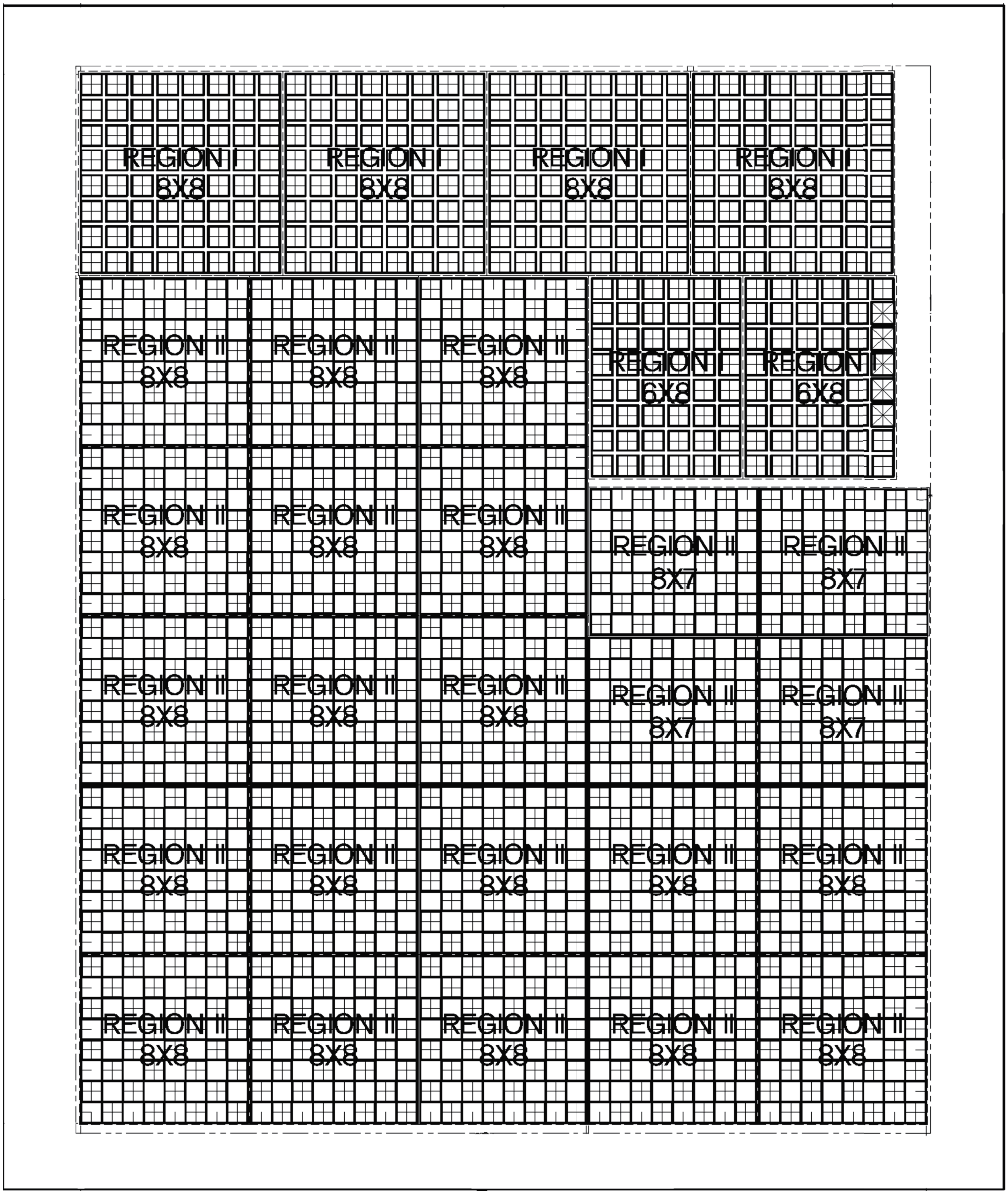

Figure 4.3-1 (page 1 of 1)
Discrete Two Region Spent Fuel Storage Rack Layout

This figure will be added to Section 4.3

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B 3.7 PLANT SYSTEMS

B 3.7.16 Spent Fuel Assembly Storage

BASES

BACKGROUND

The spent fuel storage facility is designed to store either new (non-irradiated) nuclear fuel assemblies, or burned (irradiated) fuel assemblies in a vertical configuration underwater. The spent fuel pool is sized to store 1,792 irradiated fuel assemblies, which includes storage for five failed fuel assemblies.

The spent fuel storage cells are installed in parallel rows. The center to center nominal distances between fuel assemblies for Region I and Region II are specified in Specification 4.3.1.1. This space and the neutron absorbing material attached to the storage cell are sufficient to maintain a K_{eff} less than 1.0 for spent fuel of original enrichment of up to 5 weight percent.

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APPLICABLE SAFETY ANALYSES

The spent fuel storage facility is designed for non-criticality by maintaining sufficient space and using the neutron absorbing material. The spent fuel assembly storage satisfies LCO SELECTION CRITERION 2.

LCO

The restrictions on the placement of fuel assemblies within the spent fuel pool, according to Figure 3.7.16-1 in the accompanying LCO, ensures that the K_{eff} of the spent fuel pool will always remain less than 1.0 assuming the pool to be flooded with unborated water.

The restrictions are consistent with the criticality safety analysis performed for the spent fuel pool according to Figure 3.7.16-1 in the accompanying LCO. Fuel assemblies not meeting the criteria of Figure 3.7.16-1 shall be stored in accordance with Specification 4.3.1.1.

APPLICABILITY

This LCO applies whenever any fuel assembly is stored in Region II of the spent fuel pool.

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The spent fuel storage racks are required to maintain a K_{eff} of less than 1.0 under normal conditions at a 95/95 level assuming the pool is flooded with unborated water. Compliance with this regulatory requirement has been ensured by developing storage requirements as a function of burnup and initial enrichment (Figure 3.7.16-1). Once the burnup requirements have been determined, the amount of soluble boron necessary to maintain a K_{eff} no greater than 0.95 under normal and postulated accident conditions is calculated. The details of the analyses are provided in Reference 2. It is shown that a soluble boron concentration of 1231 ppm enriched boron is required to maintain K_{eff} less than or equal to 0.95 for allowable storage configurations, which is well within the 2150 ppm enriched boron requirement of LCO 3.7.15.

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BASES

ACTIONS

A.1

Required Action A.1 is modified by a Note indicating that LCO 3.0.3 is not applied.

When the configuration of fuel assemblies stored in Region II of the spent fuel pool is not in accordance with Figure 3.7.16-1, immediate action must be taken to make the necessary fuel assembly movement to bring the configuration into compliance with Figure 3.7.16-1.

If irradiated fuel assemblies are moved while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If irradiated fuel assemblies are moved while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operation. Therefore, in either case, inability to move fuel assemblies is not sufficient reason to require a reactor shutdown.

SURVEILLANCE REQUIREMENTS

SR 3.7.16.1

This SR verifies by administrative means that the initial enrichment and burnup of the fuel assembly is in accordance with Figure 3.7.16-1 in the accompanying LCO. For fuel assemblies in the unacceptable range of Figure 3.7.16-1, performance of this SR will ensure compliance with Specification 4.3.1.1.

REFERENCES

1. DCD Tier 2, Subsection 9.1.1.

2. APR1400-Z-A-NR-14011-P, “Criticality Analysis of New and Spent Fuel Storage Racks,” KHNP, November 2014.