BWX Technologies, Inc.

May 3, 2018
18-026

ATTN: Document Control Desk
Director, Office of Nuclear Material Safety & Safeguards
U. S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Reference: (1) License No. SNM-42, Docket 70-27

(2) Letter dated December 13, 2017, Burch (BWXT NOG-L) to NRC (Document Control Desk), Request to Amend License SNM-42, Chapter 5, *Nuclear Criticality Safety*

(3) Letter dated April 3, 2018, Burch (BWXT NOG-L to NRC (Document Control Desk) Revised Request to Amend License SNM-42, Chapter 5, *Nuclear Criticality Safety*

Subject: Responses to Additional Information Request on Revised Request to Amend License SNM-42, Chapter 5, *Nuclear Criticality Safety*

Dear Sir or Madam:

In Reference 2, BWXT NOG-L requested approval for an amendment to Chapter 5 of the SNM-42 License Application in accordance with 10 CFR 70.34. In Reference 3, BWXT Nuclear Operations Group, Inc. - Lynchburg (BWXT NOG-L), submitted additional information to clarify our request to amend Chapter 5 of the SNM-42 License Application based on communication between Office of Nuclear Materials Safety and Safeguards (NMSS) staff and BWXT NOG-L. Following submittal of Reference 3 and review by NMSS, BWXT NOG-L is submitting additional information regarding the Chapter 5 amendment request based on an April 10, 2018 phone conference between Dr. Christopher Tripp (NRC), Merritt Baker (NRC) and Larry Wetzel (BWXT NOG-L). Enclosure 1 provides a summary of the requested additional information from the April 10, 2018 phone conference with BWXT NOG-L's accompanying responses.

The amendment to Chapter 5, which was submitted in Reference 3 has been modified and is included as Enclosure 2 to this letter. Enclosure 2 contains the proposed revision of Chapter 5 of the SNM-42 License Application. Proposed changes to Chapter 5 are denoted by vertical lines in the margin of the affected pages of Enclosure 2.

If you have questions or require additional information, please contact Chris Terry, Manager of Licensing and Safety Analysis, at ctterry@bwxt.com or 434-522-5202.

Sincerely,

B. Joel Burch
Vice President and General Manager
BWXT Nuclear Operations Group, Inc. – Lynchburg

Enclosures

cc: NRC, Region II
NRC, Resident Inspector
NRC, Merritt Baker

ENCLOSURE 1

Summary of NRC Requests for Additional Information (RAI) with BWXT NOG-L Responses for Amendment Request for Chapter 5 of the SNM-42 License Application

BWXT Amendment Request
Changes to Chapter 5, "Nuclear Criticality Safety"

Staff (NRC) reviewed the responses (BWXT NOG-L) to our RAIs dated April 3, 2018, with the following results. In some cases, follow-up questions are needed.

RAI 1

The response is generally acceptable. However, while the licensee has committed to annual code verification, it has removed a paragraph from LA Section 5.2.1 that had stated "When modifications are made to the computer code system (hardware or software), the impact of the change shall be assessed to determine if the system needs to be reverified." With the removal of this discussion, LA Section 5.2.1 now says, with regard to re-verification, "Verification of the computer code system will occur annually or after revision to the computer code or associated data." This no longer specifies that re-verification will take place upon changes to the computer *hardware*, which is considered part of the computer code system, only the computer code and associated data. Clarify if this is the intent, and if so justify not re-verifying upon changes to the computer hardware.

BWXT Response – RAI 1

This was unintentional. The changes to the computer code system (hardware and software) are assessed to determine if reverification is needed. This has been changed.

RAI 2

The response is acceptable.

RAI 3

The response is acceptable.

RAI 4

The response is acceptable.

RAI 5

This question was addressed by changes to LA Section 5.2.2 and 5.2.2.4. The changes are generally acceptable, but the following additional clarification is needed:

(1) Clarify the first sentence added to LA Section 5.2.2: "Interaction can be bounded for a unit if the k-effective of the system, unit of interest and an adjacent unit, under normal and credible upset conditions...". Clarify whether the intent is to define a system as a "unit of interest and an adjacent unit." If so, can this be applied to arrays and collections of units, as described in LA Section 5.2.2.4 and the example provided in your response to RAI 7?

(2) Reconcile the following apparent inconsistencies between the methods described in LA Section 5.2. and Section 5.2.2.4:

a. LA Section 5.2.2 refers to "units," whereas LA Section 5.2.2.4 refers to "units, equipment, or arrays" (see previous question about multiple units).

b. LA Section 5.2.2 refers to the adjacent unit being "removed" when the water box is modeled, whereas LA Section 5.2.2.4 states that the walls of the water box are "moved away" in making this comparison. (Note the example follows the method as described in LA Section 5.2.2.4.)

c. The last sentence in LA Section 5.2.2.4 states: "If the addition of the mobile container results in a higher k-effective than with the original water wall location, the item must be included as a normal condition." LA Section 5.2.2 refers to "normal and credible upset conditions." Clarify whether the method in LA Section 5.2.2.4 necessarily applies to the normal condition model, or if it will be applied to both the normal and credible abnormal condition models.

(3) The fourth paragraph of LA Section 5.2.2.4 contains the following: "If there is a concrete roof over the fuel, the more reactive of a 1 inch thick water slab or the concrete at its actual height and using its actual thickness shall be used. (In the case of the concrete ceiling, the water box will extend to the ceiling.) These two sentences appear to contradict each other in regard to the thickness of the water box to be modeled. Clarify what boundary conditions will be applied when there is a concrete roof over the units being modeled.

BWXT Response – RAI 5

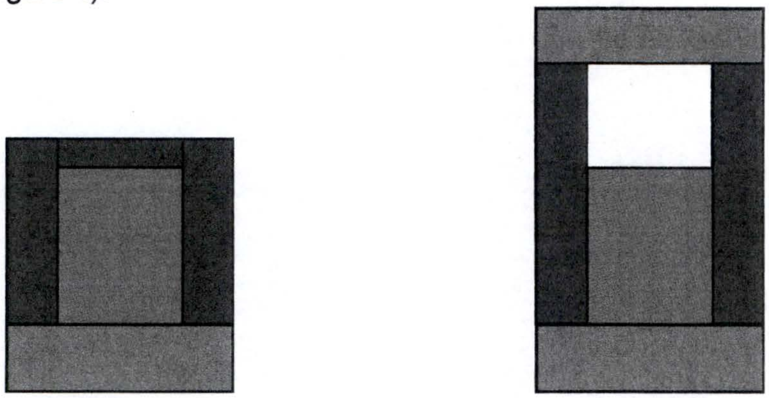
(1) Unit is a generic term what describes what is being evaluated. If the evaluation is for rack or set of columns (array) this is the unit. In KENO-V.a terms, it is the global unit. To avoid this confusion, the wording in 5.2.2 has been replaced with a pointer to methods defined later in Section 5.2.2.

(2.a) See response to (1).

(2.b) This is based on perspective. In 5.2.2, the paragraph is written starting with the two units and then removing one and including a water wall. In 5.2.2.4, the paragraph is written from the perspective of the unit being evaluated, then the water box is moved out to include the adjacent unit. The description of how isolation is demonstrated has been replaced in 5.2.2 and a pointer to the methods specified later in the section has been added.

(2.c.) The normal condition model must be established in order to properly model the upset conditions. This can be an iterative process as with the mobile container.

(3) Extending the water box means the lateral sides. To assess the concrete ceiling, the water box would be modeled with the 12 inches on the side and 1 inch directly on top. The model would then have the 1 inch water on top removed; the sides of the water box would be extended up to the concrete ceiling and the ceiling model with its actual thickness (see Figure 1).



1 inch on top **Concrete Ceiling**

Figure 1: Sketch of Models to Assess Top Reflection

RAI 6

The response is acceptable.

RAI 7

Clarify that, in the example provided and the text of LA Section 5.2.2.4, the term "base model" refers to the case where a single unit and/or array is surrounded by 12 inches of water, vs. the "interaction model" to which it is compared.

BWXT Response – RAI 7

The base model refers to what is being evaluated. The interaction model is created from the base model by moving out the side of the water box and including potential interacting items.

RAI 8

The response is acceptable.

Additional Questions

Clarify the intent in the following new material in LA Chapter 5:

(1) In LA Section 5.2.1, "analysis" is misspelled in two subscripts in the equations.

(2) In LA Section 5.2.1, in the section on non-parametric margin, in the sentence that starts "The *percent* confidence that a fraction of the population…", beta is a fraction and not a percent.

(3) In LA Section 5.2.2, the first sentence "Individual fuel units which are safe by themselves must be evaluated to determine the extent of the neutron interaction between other fuel

units in an array" appears to be missing something. Should this read "...between *them and* other fuel units in an array"? If not, clarify.

(4) In LA Section 5.2.2.4, the first sentence in the fourth paragraph appears to have an extra "is"

BWXT Response

Items 1-4 above have been corrected.

ENCLOSURE 2

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SNM-42

CHAPTER 5

NUCLEAR CRITICALITY SAFETY

CHAPTER 5
NUCLEAR CRITICALITY SAFETY
TABLE OF CONTENTS

5.1 Nuclear Criticality Safety Specifications

The Manager of Nuclear Criticality Safety (NCS) has the overall authority and responsibility for the implementation of the Nuclear Criticality Safety program for the site. The manager's authority includes terminating any operation deemed to be unsafe or contrary to license conditions, or contrary to good safety practice. The manager's responsibilities include: maintaining "state-of-the-art" computational methods and practices, determining the need for Nuclear Criticality Safety evaluations, performing evaluations, and preparing Nuclear Criticality Safety postings, ensuring they are properly posted to guide safe operations, and maintaining Nuclear Criticality Safety inspection and audit programs for the plant. The Manager of Nuclear Criticality Safety is responsible for training the NCS staff to perform their duties. Oversight of the Specialized Nuclear Criticality Safety Training Program and the NCS portion of General Employee Safety Training, as described in paragraph 5.1.4 of this chapter, is also the responsibility of the Manager, Nuclear Criticality Safety.

However, these responsibilities do not relieve area management of their responsibility for ensuring that operations are conducted in compliance with Nuclear Criticality Safety requirements. Decisions of the Manager, Nuclear Criticality Safety are not to be compromised by direct pressures of time or production.

The decision to perform a Nuclear Criticality Safety evaluation is based on the need to establish new or revised Nuclear Criticality Safety limits. Requests for a Nuclear Criticality Safety evaluation can originate from supervisors, managers, and engineers who are knowledgeable of the process or equipment changes.

5.1.1 Protection Against Criticality

The site is committed to implementing the following NCS program objectives:

(a) preventing an inadvertent nuclear criticality,

(b) protecting against the occurrence of an identified accident sequence in the ISA Summary that could lead to an inadvertent nuclear criticality,

(c) complying with the NCS performance requirements of 10 CFR 70.61,

(d) establishing and maintaining NCS safety parameters and procedures,

(e) establishing and maintaining NCS safety limits and NCS operating limits for IROFS,

(f) conducting NCS evaluations to assure that under normal and credible abnormal conditions, all nuclear processes will remain subcritical and maintain an approved margin of subcriticality for safety,

(g) establishing and maintaining NCS IROFS, based on current NCS evaluations,

(h) providing training in emergency procedures in response to an inadvertent nuclear criticality as described in Chapter 8,

(i) complying with NCS baseline design criteria requirements in 10 CFR 70.64(a) as described in Chapter 11,

(j) complying with the NCS ISA Summary requirements in 10 CFR 70.65(b) as described in Chapter 3, and
(k) complying with the NCS ISA Summary change process requirements in 10 CFR 70.72 as described in Chapter 11.

The site is also committed to the following double contingency policy: "Process designs shall incorporate sufficient factors of safety to require at least two unlikely, independent, and concurrent changes in process conditions before a criticality accident is possible." This statement of the double contingency policy is a modification of the policy as defined in ANSI/ANS-8.1-1998 in that the policy is mandatory. There are, however, systems which cannot feasibly utilize classical double contingency protection. Nuclear Criticality Safety for such systems is assured through defense-in-depth to prevent unwanted changes in any one process condition that might adversely affect system safety. Defense-in-depth utilizes two or more reliable barriers or controls to protect against such unwanted changes. Defense-in-depth is enhanced through diversity and redundancy of barriers and controls. The barriers or controls used for defense-in-depth shall be reviewed to ensure that they are not subject to common mode failure (i.e., one malfunction could lead to loss of two barriers or controls). Control of two parameters is preferred over multiple controls on a single parameter. Any deviation from the Double Contingency Principle or of a defense-in-depth protection where Double Contingency is not feasible shall require approval of the Manager of Nuclear Criticality, the Change Review Board, and the U. S. Nuclear Regulatory Commission (via a license amendment).

5.1.2 Nuclear Criticality Safety Procedures and Postings

Activities at the site involving special nuclear material are conducted according to limits and controls established by Nuclear Criticality Safety. The administrative limits and controls are provided to the operating areas on Nuclear Criticality Safety postings or in operating procedures or both. Engineered limits and controls are provided in operating and maintenance procedures as necessary.

Nuclear Criticality Safety postings shall describe the administrative limits and controls for a particular area, operation, work station, or storage location as appropriate for providing workers a ready reference for verifying compliance and safe operation. Nuclear Criticality Safety limits and controls are posted according to procedural requirements and instructions maintained by Nuclear Criticality Safety.

Nuclear Criticality Safety postings will include the following information as a minimum:

- Type of material permitted.
- Form of material.
- Allowable quantity (number of containers, pieces, weight, or volume).
- Spacing of fuel units, if required.

- Restriction on the presence of moderators, if required.

Storage vessels such as cans, buckets, etc., which contain special nuclear material will be labeled as to the type and amount of material. In-process material, i.e., materials being processed for use in a finished product, and scrap (10 CFR 74.4) will be handled with knowledge of type and quantity of material whenever practicable. When the type or quantity is not known, such material shall be handled in favorable geometry or volume until the material can be assayed. Dry Waste material that is contaminated with low levels of uranium may be classified by operating personnel as Dry Low Level Waste in accordance with written guidelines as established in a site-wide Quality Work Instruction. Dry Low Level Waste may be collected in appropriately labeled 55-gallon type containers. Fifty-five gallon containers to which Dry Low Level Waste has been added during any day, shall be assayed for U-235 content on that day or at a frequency approved by the Nuclear Criticality Safety Manager and specified in the above site-wide Quality Work Instruction. The U-235 content of 55-gallon Dry Low Level Waste containers shall not exceed 300 grams.

5.1.3 Nuclear Criticality Safety Audits and Inspections

NCS inspection of selected site operations involving SNM, shall be performed weekly by NCS Engineers to determine if activities are being conducted in accordance with Nuclear Criticality Safety limits. Inspections will be performed at least monthly on selected weekends or back shifts. Additionally, Radiation Control Technicians shall perform daily inspections in unencapsulated fuel handling areas that are in operation.

NCS audits of selected plant activities involving SNM shall be conducted quarterly. Audits shall be conducted by a Nuclear Criticality Safety Engineer. The entire site, where SNM is processed or stored, shall be audited biannually.

The purposes of the audits are:

- To determine that site operations are conducted in compliance with the NCS aspects of regulatory requirements, license conditions, operating procedures, and posted limits.
- To determine the adequacy of administrative controls and postings and to verify the use of sound NCS practices.
- To examine equipment and operations to determine that past evaluations remain adequate.
- To examine trends in findings of NCS inspections and the adequacy of corrective actions.

5.1.4 Nuclear Criticality Safety Training

5.1.4.1 General Employee Safety Training

All individuals are given nuclear criticality safety training prior to being granted unescorted access to the Restricted Areas as defined by 10 CFR 20. This includes, as a minimum, the following training:

- A discussion about the fission process and criticality.
- A brief history of criticality accidents.
- The effects and consequences of a criticality accident at this plant.
- The importance of an immediate evacuation in case of a criticality accident.
- A discussion about the basic nuclear criticality safety controls used at NOG together with appropriate examples of the various controls.
- A discussion about the nuclear criticality safety postings, i.e., signs.
- A discussion about nuclear safety violations and the impact they have on the nuclear criticality safety program.

This training shall be developed by the Training Specialist with the technical oversight of Nuclear Criticality Safety. Expertise from various areas of the company as well as outside the company may be used in the development of this training program. This training is repeated annually. Its development and presentation is done according to approved procedures.

5.1.4.2 Specialized Instruction

In addition to General Employee Safety Training, all employees who handle fissile materials are given specialized instruction annually. This program covers the general safety principles of handling fissile material and also covers the application of these principles by discussing examples of specific criticality safety limits. Specialized Nuclear Criticality Safety training shall be developed by the Training Specialist with the technical oversight of Nuclear Criticality Safety. Expertise from various areas in and outside the company may be used in the development of this training.

Specialized training is supplemented by on-the-job training and qualification of operators. This training specifically addresses the criticality safety limits contained in operating procedures and on postings for specific jobs. In addition, a new operator will work with an experienced operator until the supervisor judges that the new operator understands the safety requirements well enough to perform the job alone.

5.1.4.3 Evaluation of Training

The effectiveness of the Nuclear Criticality Safety training is judged by three methods.

First, written and/or oral tests are given each individual who receives Specialized Nuclear Criticality Safety instruction; the test must be passed. Tests are not normally given following General Employee Safety Training.

Second, Nuclear Criticality Safety inspections of the entire plant reveal how well personnel understand the safety controls as a function of the number of Nuclear Criticality Safety violations found.

A third method of evaluating how well employees understand the safety requirements is the supervisor's close contact with the employee. Through discussions and job performance appraisals, a supervisor is well informed to determine if an employee understands the Nuclear Criticality Safety limits. As the supervisor thinks necessary, an employee may be retrained to the point where his supervisor's confidence in him is raised to an acceptable level.

5.1.5 Nuclear Criticality Monitoring System

The site shall maintain a nuclear criticality monitoring system for each area in which 700 grams or more of U-235 is possessed, 450 grams or more of plutonium, or 450 grams or more of any combination thereof. This monitoring system shall be capable of energizing a clearly audible alarm signal if accidental criticality occurs. The placement of the detectors shall be determined by calculation utilizing detection criteria described in 10 CFR 70.24(a)(1), and methodology described in Regulatory Guide 3.71, Nuclear Criticality Safety Standards for Fuels and Material Facilities, Revision 2, December 2010.

Whenever the criticality monitoring system is out of service, in storm-watch mode, or being tested or repaired, compensatory measures shall be in place to ensure evacuation if a criticality occurs. Compensatory measures shall be specified in facility procedures, and periods when the criticality monitoring system is out of service should be minimized to the extent practical.

5.1.6 Liquid Effluent Monitoring

Liquid waste from plant areas processing unclad uranium drains through one of two identical monitoring systems, and collects in retention tanks prior to transfer to the Waste Treatment facility. One system is called the "Recovery" system, and the other is identified as the "Plant" system. The Recovery system drains liquids from Bays 13A, 14A, 15A, Bay 14A exhaust scrubber, and the Compactor Area. All other areas in which unclad uranium is processed are drained through the Plant system. The contaminated liquid waste lines are of a favorable geometry. These lines are monitored by an in-line monitor that has a preset alarm level not greater than 0.04 g U-235/liter, which is sufficient to assure that Nuclear Criticality Safety limits are not exceeded. In addition to the sounding of an alarm, the high concentration activates an automatic valve in either line which immediately shuts off flow of solution. Notification of alarm conditions and responsive actions are performed according to approved procedure or under the direction of the Emergency Response Organization.

Each in-line monitor at the retention tank site is calibrated biannually according to approved procedures. The electronics of each monitor are checked weekly when the drained plant areas are in operation, against a radiation source to ensure proper operation. Records are kept of the biannual calibration and the weekly equipment check.

Each of the retention tanks is inspected monthly for sludge buildup when the tank has been in service. The solution that enters these tanks is acidic which retards precipitation. In addition, routine fluid agitating and tank flushing procedures prevent sludge accumulation in the bottom of the tanks.

The pit basin beneath the retention tanks may be utilized to collect additional effluent if the volume afforded by the retention tanks is insufficient. Liquid in the pit is pumped through the measurement system and processed at the Waste Treatment Facility.

5.2 Nuclear Criticality Safety Criteria

The design of equipment and establishment of operating safety limits shall consider the pertinent process conditions and known modes of failure. The most credible combination of the fissile material density, H/X ratio, solution concentration, reflection, interaction, interspersed moderation, and measurement uncertainty are assumed before Nuclear Criticality Safety limits are established. Certain conditions may be deemed incredible if specifically excluded by experimental evidence or design considerations; such as, experimental data showing maximum densities achievable for certain compounds. The Change Review Board reviews and approves new or proposed changes in which there is a Nuclear Criticality Safety element not previously reviewed for the specific operation.

The Nuclear Criticality Safety limits are established by using a variety of techniques which are described below. Current design criteria used to ensure Nuclear Criticality Safety are described in detail in the Appendix to this chapter. Design philosophy and criteria have evolved over the years to require a more structured and formal approach to Nuclear Criticality Safety analyses. The current design criteria of the Appendix are the culmination of this evolutionary process and have been in use since January 1, 1995.

5.2.1 Computer Codes

Computer programs are used extensively to model the many processes required to manufacture the reactor cores. The analysis of the calculations leads to the established Nuclear Criticality Safety limits. Nuclear Criticality Safety calculations are done using well benchmarked and verified computer codes, techniques and cross section data sets. The computer codes are run on controlled software and hardware. In addition, it is anticipated that new techniques and cross section information will from time to time be developed; these techniques may be incorporated into the analyses when they have been properly benchmarked.

Application of computer codes for Nuclear Criticality Safety analysis is under the direction of a qualified Nuclear Criticality Safety engineer, who ensures that the pertinent computer codes, assumptions, and techniques are properly used. The qualified Nuclear Criticality Safety engineer also ensures that computer codes are applied to the appropriate Area(s) of Applicability or provide justification for applying it outside the Area(s) of Applicability. All such analyses are then reviewed by an independent qualified individual. Neither the analyzing or reviewing engineer has direct responsibility for the manufacturing operation to be performed, and he does not report to persons who are immediately responsible for such operations.

Validation of computer codes and cross section sets is done per ANSI/ANS-8.24-2017. A computer code will be verified prior to validation. Verification of the computer code system will occur annually or after revision to the computer code system. The application of new computer codes or additional benchmark data will be reviewed and approved by an independent reviewer meeting the minimum qualifications of a Nuclear Criticality Safety engineer.

Calculational Margin and Margin of Subcriticality:

The calculational margin includes the allowances for the bias (calculational k-effective minus the experimental k-effective value or the ratio of the calculational k-effective to the experimental k-effective minus 1) and the bias uncertainty as well as uncertainties associated with interpolation, extrapolation and trending. The bias uncertainty accounts for the uncertainties in benchmarks, the calculational models and the calculational method. An acceptable k_{eff} is determined by:

$$1 - \Delta k_{MoS} - \Delta k_{CM} \geq k_{calc(analysis)} + 2\,\sigma_{calc(analysis)}$$

$$(USL = 1 - \Delta k_{MoS} - \Delta k_{CM})$$

The preferred form is:

$$1 - \Delta k_{MoS} \geq \Delta k_{CM} + k_{calc(analysis)} + 2\,\sigma_{calc(analysis)}$$

where: Δk_{MoS} is the Margin of Subcriticality (MoS – listed in Section 5.2.3),
Δk_{CM} is the Calculational Margin,
$k_{calc(analysis)}$ is the calculated k-effective of a system being evaluated as part of Nuclear Criticality Safety analysis,
$\sigma_{calc(anlaysis)}$ is the uncertainty on that calculation.

Techniques for Establishing the Calculational Margin:

The Non-Parametric Method (NPM) is primarily applied when the underlying distribution of the data is not known or cannot be verified. The confidence level that a fraction of the population is above an observed value is

$$\beta = 1 - \sum_{j=0}^{m-1} \frac{n!}{j!(n-j)!}(1-q)^j q^{n-j}$$

where:

q is the desired population fraction (0.95)
n is the number of benchmarks in the data set
m is the rank order indexing from the smallest sample to the largest (m=1 for the smallest sample, m=2 for the second smallest, etc.).

For the smallest value in the sample, the equation reduces to

$$\beta = 1 - q^n = 1 - 0.95^n$$

If there are more than 58 benchmarks, a rank order (m) greater than 1 can be used provided the selected rank order yields a β equal to or greater than 0.95 which would assure at least a 95/95 confidence.

Non-parametric methods are the preferred method to establish the calculational margin. The non-parametric approach used is based onthe specified rank order calculated k-effective value. This method has three terms that define the calculational margin (Δk_{CM}).

$$\Delta k_{CM} = |bias| + \sigma_{calc} + \Delta k_{NPM}$$

where: bias = k_{calc}-k_{exp} , or k_{calc}/k_{exp} - 1
k_{calc} used is the specifiedrank order calculated k-effective of the benchmarks used for the validation,
k_{exp} is the reported experimental k-effective for the same configuration,
k_{NPM} is the margin accounting for the amount of experimental data.

Since no credit is taken for a positive bias, if the specified rank order calculated k-effective of the benchmarks is greater than the experimental value, the bias is set to zero and the equation becomes:

$$\Delta k_{CM} = 0 + \sigma_{calc} + \Delta k_{NPM} \text{ (for } k_{exp} \leq k_{calc}\text{)}$$

The non-parametric margin (Δk_{NPM}) is based on the degree of confidence for 95% of the population and is shown below.

Degree of Confidence for 95% of the population (β)	**Number of Experiments (n)**	**Non-Parametric Margin (Δk_{NPM})**
>0.9	n > 45	0.00
>0.8	32 < n < 44	0.01
>0.7	24 < n < 31	0.02
>0.6	18 < n < 23	0.03
>0.5	14 < n < 17	0.04
>0.4	10 < n < 13	0.05
<0.4	n < 10	Insufficient data

β - Percent confidence that a fraction of the population is above the lowest point.

Other statistical methods such as Lower Tolerance Band (95/95 or greater) or Lower Tolerance Limit (95/95 or greater) may be used if data meets the assumptions of the methodology. When methods that employ trending are used, trends may predict k-effective values greater than unity for some parameter ranges. In ranges where the trended k-effective value exceeds unity, additional margin shall be applied equal to the amount of the positive bias as a function of the trending parameter. For methods that use average values, it is possible to have average k-effectives that exceed unity. In those cases, the additional margin shall be applied. The margin shall be the amount of the positive bias.

<u>Area of Applicability:</u>

The Area of Applicability covers processes involving: 1) unclad fuel under mass/moderator limits or volume limits such as 2.5 liter containers in many different operations like gloveboxes, storage racks, and vaults, 2) uranium-bearing solutions which are handled as product or waste, and 3) clad fuel components. The Area of Applicability includes entire range of enrichments, multiple fuel forms, and reflector/poison materials. If extensions to the Area of Applicability are required, they will be made consistent with assumptions and limitations of the method used to establish the calculational margin.

5.2.2 Neutron Interaction

Individual fuel units which are safe by themselves must be evaluated to determine the extent of the neutron interaction with adjacent fuel units. Methods of evaluating the extent of the neutron interaction are described later in this section.

A unit containing fuel may be considered isolated from another unit if the separation (edge-to-edge of fuel) is greater than the larger of the following distances:

a. twelve feet, or

b. The greatest distance across an orthographic projection of either array on a plane perpendicular to a line joining their centers.

Units may also be separated by twelve inches of concrete with density of at least 140 lb/ft^3 provided that the unit or units cannot be representable as a slab which interacts with other SNM primarily through its major face.

Computer codes validated in accordance with Section 5.2.1 can be used to evaluate the neutron interaction between individual subcritical units. These codes may be used for this purpose if they are believed to be more appropriate than the semi-empirical methods described below. Before being used for this purpose the codes will be benchmarked against appropriate critical array data.

5.2.2.1 The Solid Angle Technique

The solid angle method is used to specify safe parameters, that is, the spacing and number of units for an array without prior determination of the array multiplication factor. The set of rules is based on the assumption that the k_{eff} of an array is determined by the values of k_{eff} of the individual units and by the probability that neutrons leaking from one unit will be intercepted by another. That probability is related to the total solid angle subtended at a unit by the other units of the array. "Shadowing" has not been considered in the solid angle analysis. This makes the calculation conservative.

This method has been correlated with extensive experimental results for many different arrays of variously shaped units containing U-235 in a variety of forms. The solid angle method is used for array analysis of homogeneous low density oxides solutions and UNH solutions.

This method will be used in accordance with TID-7016, Revision 2.

5.2.2.2 The Lattice Density Technique

The lattice density model shall be applied in accordance with TID-7016, Revision 1, pages 25 through 28, as revised. Other units may be substituted for these given values, or demonstrated by a properly benchmarked computer calculation, to be less reactive when fully reflected than those given in Table IV of TID-7016, Revision 1.

5.2.2.3 The "Law of Substitution"

The "law of substitution" states that units of array "A" can be intermixed with units of array "B", provided the calculated k-effective of both arrays individually are equal to or less than that permitted by Paragraph 5.2.3. The comparison requires that both arrays be calculated in an infinite planar array. The infinite planar array is infinite in the x-y direction parallel to the floor and finite in the z direction. The z direction must be the actual array height.

In addition, the k_{eff} calculations of Arrays "A" and "B" shall include optimum interspersed moderation and any credible internal moisture or plastic mixed with the fuel. The evaluation will determine the minimum acceptable edge to edge spacing between the units in each array. When the spacing between the units in arrays "A" and "B" are different, the largest edge to edge spacing will apply to the array of mixed units. Each unit in the array must be separated by a minimum of 12 inches edge to edge. Interspersed moderation shall be considered between the units in the infinite planar array. Credible reflectors shall also be considered in the array.

The k-effective of the mixed array will not exceed the infinite planar array of array "A" or array "B". Arrays "A" and "B" shall also satisfy the requirements of both 5.1.1 and 5.2.3.

5.2.2.4 Water Box

The "Water Box" method uses thick water reflectors to bound the interaction with adjacent fuel units and reflection from adjacent equipment and structures.

The lateral sides of fuel units, equipment or arrays shall be modeled as a 12 inch thick water box. The water box will extend to the top of the fuel. If the fuel is or could be within 12 inches of a concrete wall, the associated side of the water box shall be modeled as 12 inch thick concrete or 12 inch thick water, whichever is more reactive. The floor shall be modeled as 12 inch thick concrete.

Hand-carried fuel or fuel container shall include a water wrap that is equivalent of human hands. For groups of containers, the analysis shall use the more reactive of the individual containers with a "hand" wrap or the group of containers with the "hand" wrap. The "hand" wrap for a group of containers shall be modeled as a close-fitting cylinder or rectangle depending on which is more reactive.

If there is a drop ceiling and/or metal and tar roof above the fuel, a 1 inch thick water slab shall be placed on the top. If there is a concrete roof over the fuel, the more reactive of a 1 inch thick water slab or the concrete at its actual height and using its actual thickness shall be used. (In the case of the concrete ceiling, the water box will extend to the ceiling.

If there are fuel operations on the floor above, explicit calculations must be used to assess interaction. The explicit calculations may be used to establish a bounding top reflector condition.

To assess whether the water box bounds interaction, the walls of the water box are moved away from the base unit and adjacent units are included in the model. If the k-effective does not exceed that of the base unit, the water box bounds interaction. If the k-effective does exceed the base unit, the base and adjacent unit must be modeled together as the normal condition. This assessment must address any fuel unit within 12 feet of the base unit or the greatest distance across an orthographic projection of either array on a plane perpendicular to a line joining their centers, whichever is larger.

For assessing the mobile containers, one of the water walls is moved far enough away from the unit to allow the additional item to be placed into the model. If the addition of the mobile container results in a higher k-effective than with the original water wall location, the item must be included as a normal condition.

5.2.3 Nuclear Criticality Safety Limits

Since there is no general correlation between k-effective and variations in physical parameters except at the point of criticality (k-effective = 1), the safety of operations where reactivity is calculated is based on an understanding of the safety margins provided by controlled parameters. For each controlled parameter, a determination is made of the correlation between k-effective and variations in the controlled parameter. This correlation along with an assessment of the measurement uncertainty for the controlled parameter and the ability to detect and control process variations that affect the controlled parameter is used to establish adequate safety margins. This approach shifts the focus from an arbitrary k-effective value as an indication of the available safety margin to an understanding of the sensitivity of the k-effective to changes in controlled parameters.

For each controlled parameter, the values of the parameter that correspond to the Failure and Safety Limits are determined. The Failure Limit is defined as the point at which the system is critical; its k-effective value therefore is 1. The Safety Limit is set below the Failure Limit value as an added margin of safety. The k-effective for the Safety Limit shall not exceed:

- 0.97 [$\Delta k_{MoS} = 0.03$] for low-enriched systems (uranium enriched ≤ 10 weight percent in U^{235}),
- 0.975 [$\Delta k_{MoS} = 0.025$] for systems involving welded Naval Reactors clusters in which the welded Naval Reactors cluster is the reactivity driver of the system, and
- 0.95 [$\Delta k_{MoS} = 0.05$] for all other high-enriched systems (uranium enriched > 10 weight percent in U^{235}).

The k-effective value for the Safety Limit for low enriched systems is 0.97, because low enriched systems are less sensitive to changes in parameters affecting reactivity than are high enriched systems. For example, 400 grams of fully enriched uranium is required to increase the k-effective of a water reflected sphere of uranium metal (400 grams U/l) and water from 0.95 to 1.0, but 37,000 grams of uranium must be added for 5 w/o material. For the same uranium metal and water solutions, an infinite water reflected cylinder diameter increases by 1.1 cm as k-effective increases from 0.95 to 1.0 for fully enriched material, but the same system would have to increase by 12.8 cm if the material were 5 w/o. These are just two generic examples which demonstrate the low sensitivity of low enriched systems to changes in reactivity parameters. The Limiting Condition of Operation (LCO) value shall be determined after consideration of credible accident scenarios consistent with the double contingency principle. The LCO value is set such that any single failure (contingency) in the controlled parameter will not exceed the Safety Limit value for that parameter. Also, the k-effective value for the LCO shall not exceed:

- 0.94 [$\Delta k_{MoS} = 0.06$] for low-enriched systems (uranium enriched ≤ 10 weight percent in U^{235}),
- 0.94 [$\Delta k_{MoS} = 0.06$] for systems involving welded Naval Reactors clusters in which the welded Naval Reactors cluster is the reactivity driver of the system, and
- 0.92 [$\Delta k_{MoS} = 0.08$] for all other high-enriched systems (uranium enriched > 10 weight percent in U^{235}).

In order to use the higher k_{eff} limit, a NR welded cluster must have the following attributes:

- Be fueled by high enriched uranium (>90 weight percent ^{235}U).
- Have a thermal neutron spectra when full flooded.
- Be constructed of the same geometric style elements as those in the KAPL critical experiments.
- Any significant absorbers must have been included in the KAPL critical experiments.

The basis for applying the higher k_{eff} limit to any NR welded cluster shall be documented.

A Routine Operating Limit (ROL) shall be established from the LCO to account for measurement uncertainties and normal process variability. The ROL value shall not exceed the LCO value. Additional discussion of the four limits can be found in the appendix to this chapter.

In addition to established k-effective limit for the LCO, the LCO establishes a safety margin based on measurable quantities (controlled parameters). For a controlled parameter to exceed its LCO value, a contingency would have to occur. After this contingency, the k-effective of the system would have to be less than the Safety Limit. In other words, no single contingency will take a system critical or even above its safety limit. Calculated k-effective values shall include appropriate allowances for any bias in data and calculational methods used.

5.2.4 Nuclear Criticality Safety of Individual SNM Units

Critical data from experiments are available from handbooks, published papers, and other documents. These documents can be used by a qualified Nuclear Criticality Safety engineer to determine the critical dimensions, concentrations, etc., for various types, shapes, and sizes of units. Nuclear Criticality Safety limits for individual SNM Units can then be derived using handbook data, provided an adequate safety factor is applied that is at least equal to the safety of calculational methods. However, the use of handbooks and hand calculations is not always adequate, in many cases, to determine the k-effective of the individual fuel unit or an array of units. Calculational methods, including use of an approved margin of subcriticality, may also be necessary to determine the critical dimensions, concentrations, etc., for various types, shapes, and sizes of units.

5.2.5 Safety Factors

Safety factors are applied to single isolated units containing fissile material. These factors, given below, reduce the critical dimension, critical volume, critical mass, and critical concentration to assure the unit is subcritical.

Calculations to determine the accident condition k-effective is based on optimum moderation unless moderating materials such as polyethylene, water, and paper are restricted or carefully controlled. In addition, calculations for all degrees of interspersed moderation may not be necessary if it can be demonstrated that certain degrees of interspersed moderation between fuel units is not a credible accident. Individual containers are considered to be moderation controlled if the following criteria are met:

5.2.5.1 The structural integrity of the container is such that it prevents the inadvertent entry of water, e.g., a metal can.

5.2.5.2 The container is used in a manner such that water could not inadvertently enter the container, e.g., the lid or top of the container is in place.

5.2.5.3 The H/X of the contents of the container is calculated if any moderating material is in the container. The H/X will not exceed the maximum H/X on which the mass limit of the container is based. Moderation controlled areas are so called because large amounts of moderating materials are eliminated or carefully controlled. Normally, all water and steam lines are either left out by design, disconnected and plugged, double cased, or shielded. Fire fighting procedures for moderation controlled areas preclude the use of water. Plastic, oil, and other hydrocarbon materials are carefully controlled in insignificant quantities. The amount of water and plastic that goes into the containers is controlled. The H/X is calculated for each container and the amount of U-235 is also known; therefore, Nuclear Criticality Safety limits for the storage locations such as 10 kilograms, H/X ≤ 2 and 3.6 kilograms, H/X ≤ 20 - are confirmed.

When nuclear criticality calculations indicate that license conditions would be violated at a certain degree of interspersed moderation other than 100%, then all degrees of moderation exceeding any allowed H/X limit must be controlled. In the same respect, if ranges of interspersed moderation are determined not to be credible or a moderation control area is required, then all degrees of interspersed moderation exceeding the H/X limit, or being outside the credible range of interspersed moderation, are excluded from the criticality calculations. For example, in the Central Storage Vault, there are no water or steam lines to break; the building is situated on a hill approximately 75 feet above the highest recorded flood level of the James River; the nearest floor drains are in an adjacent area and even if they back up, the water would run out under the doors; fire fighting procedures and a posted sign preclude the use of water. Interspersed moderation exceeding the H/X limit is virtually impossible.

When Nuclear Criticality Safety requires the exclusion of moderating materials, a control system shall be used to ensure such exclusion. The control system shall include elements of training, procedures, and postings. Physical controls shall be used when appropriate.

However, whenever practical, equipment is designed by physically limiting the dimensions so criticality cannot be achieved under any foreseeable conditions. For material limited by dimension, the dimension will not exceed 90% of the critical dimension for cylinder diameters and 85% of the critical dimension for slab thicknesses. When a unit is limited by volume, the maximum allowed value will not exceed 75% of the critical spherical volume.

When it is not possible to control a physical dimension of a vessel so the unit is geometrically favorable, then reliance is placed on controlling some other parameter such as mass or concentration. For accumulations limited by mass, the maximum permissible mass will not exceed 45 percent of the critical mass if double batching is credible or 75 percent of the critical mass, if it is not credible. Provision for the largest batch size is considered when double batching is not credible. If the safety depends solely on control of concentration, then the maximum concentration is no more than 45 percent of the minimum critical concentration at optimum moderation for minimum critical concentration.

Posted Nuclear Criticality Safety limits are presented in a simple, concise, straight-forward manner such that confusion is minimized. In addition, Nuclear Criticality Safety considerations shall be an integral part of any process design. This will enhance the effectiveness of the Nuclear Criticality Safety program by establishing "built-in" Nuclear Criticality Safety controls.

When criticality control is dependent upon structural integrity to position special nuclear material, the design will include an adequate strength factor to assure against failure under foreseeable loads and accident conditions. Favorable geometry equipment shall be checked for proper dimensions and/or volume prior to being released for use. The results are documented and maintained on file. The structural integrity of units and framework of an array are checked by having the first unit manufactured from an original design fully weight-tested. Thereafter, the integrity of all units of that design is assured by manufacturing integrity. This type of weight-testing and control is not applied where the rack or unit is an integral part of the structure of the facility. Facility integrity then becomes the assurance. Lifting devices utilized to perform major lifts of large SNM bearing components fabricated for the Naval Reactors program, shall meet the design, maintenance, and operator qualification requirements of General & Administrative Requirements (GAR), Section 5.15. Devices used to ensure the physical spacing between fissile units shall not be altered without the prior approval of Nuclear Criticality Safety.

Neutron poisons that are used as the primary Nuclear Criticality Safety control of a system are certified by the manufacturer to be as ordered. Copies of the certification are kept on file. Poison material when used in a fixture is verified to be of proper dimensions and in the proper locations prior to final welding. Once it is welded in place, it is incredible that poison material could be removed inadvertently by mechanical means. Poison fixtures are checked frequently by manufacturing personnel and over-checked periodically by Nuclear Criticality Safety. When solid neutron poisons are used to ensure the Nuclear Criticality Safety of a component or system, they shall not be removed from the system or component without prior approval.

Poison fixtures that are subject to hostile environments are periodically inspected per written procedure. Fixtures that are not subject to hostile environments are not routinely inspected.

5.2.6 Unirradiated Commercial (PWR and BWR) Fuel Assemblies at LTC

Unirradiated fuel assemblies will be received at a maximum of two at a time in a shipping container licensed for two assemblies, or one assembly in a shipping container licensed for one assembly. Unirradiated fuel assemblies may be handled and stored subject to the following conditions:

5.2.6.1 Unirradiated fuel assemblies may be stored in air in the Cask Handling Area (CHA) or in the Development Test Area.

5.2.6.2 Assemblies may be stored in their shipping container as received.

5.2.6.3 Assemblies may be stored a minimum of 21-inches apart surface-to-surface.

5.2.6.4 Assemblies may be stored under water in the CHA pool, Pool Test Facility pool, or Development Test Area pool in racks constructed to maintain a 1-foot minimum surface-to-surface separation between assemblies and any other SNM. Assemblies may be handled and dismantled under water subject to the same requirements of the irradiated fuel in the CHA Pool.

5.2.6.5 No more than four unirradiated assemblies may be kept at the LTC site at one time.

5.2.6.6 Only one unirradiated fuel assembly shall be dismantled or reassembled at a time in the Development Test Area. The dismantling operation shall meet the following:

- Only one fuel rod may be removed from or inserted into the assembly at a time.
- Only one fuel rod may be in transit to any location at a time.
- The fuel assembly may be completely disassembled by withdrawing one fuel rod at a time from the assembly; during all stages of disassembly, the partially disassembled assembly shall be maintained within the confines of the assembly whether damaged or undamaged.

5.2.6.7 Associated with the dismantling operation, one storage position will be permitted for fuel rods removed from an assembly provided that:

- The assembly and associated rod storage position shall be separated from each other and from any other fissile material by a minimum of 21 inches surface-to-surface.
- The associated rod storage position shall be no larger in any dimension than the fuel assembly. There shall be one storage position for each fuel assembly to be dismantled. Rods may be stored or handled in a slab up to 4 inches thick provided the slab is separated from other fissile material by a minimum of 12 feet.
- Only one fuel rod may be removed or inserted into the associated rod storage position at a time.

5.2.6.8 Fuel assemblies to be studied shall meet the following:

- Each assembly shall be of the enriched PWR type with a 15 x 15, or 17 x 17 square pin lattice not greater than 8.6 inches on a side (further identified as Babcock & Wilcox Mark B or Mark C canless assemblies).
- The maximum initial enrichment in an unirradiated fuel assembly shall not exceed 4.05 wt%.
- Damaged fuel assemblies may be examined in air. Fuel assemblies which have been damaged can be examined in water if they maintain their 8.6 inch on a side dimension.

5.2.6.9 Other PWR or BWR fuel assemblies which do not meet the above listed requirements may be studied, provided:

- The unirradiated, fully reflected fuel assembly (fueled with UO2 only) with all control rods removed is shown by an appropriate Nuclear Criticality Safety evaluation to be subcritical by at least 5% (k-effective < 0.95).
- The fuel assembly is shown by an appropriate safety evaluation to be subcritical by at least 5% k-effective < 0.95) under specific conditions of disassembly.

5.2.6.10 BWR fuel assemblies may be received and studied provided:

- They are evaluated pursuant to the previous section (5.2.6.9), or
- The BWR fuel assemblies have a maximum initial unirradiated enrichment of 4.05 wt% U-235 and have a cross sectional area not exceeding that of a 22.5 cm (8.85 in.) diameter cylinder.

5.2.7 Irradiated Commercial (PWR and BWR) Fuel Assemblies at LTC

Irradiated fuel assemblies will be received at a maximum of two at a time in a shipping container licensed for two assemblies, or one assembly in a shipping container licensed for one assembly.

5.2.7.1 Irradiated fuel assemblies will be stored in the CHA pool which is limited to the following conditions:

- A maximum of four fuel assemblies or portions thereof may be in the pool at a time.
- The assemblies shall be stored in racks constructed to maintain a 1-foot minimum surface-to-surface separation between assemblies and any other SNM in storage or transit. Each position in the assembly storage rack must limit contained fuel to a square not to exceed the dimensions of a fresh fuel assembly or to a cross sectional area not exceeding that of a 22.5 cm (8.85 in.) diameter cylinder.
- Partially dismantled assemblies will be stored in the assembly storage rack.
- Only one assembly may be in a designated work area of the pool at any one time. There shall be a minimum of 1 foot surface-to-surface separation between the assembly in the work area and any other fissile material.

5.2.7.2 Dismantling of irradiated fuel assemblies is permitted in the Pool under Hot Cell No. 1 provided:

- Only one fuel rod at a time shall be removed from or inserted into the fuel assembly
- A fuel assembly can be completely dismantled by withdrawing one fuel rod at a time from the assembly; during all stages of dismantlement, the partially dismantled assembly shall be maintained within the confines of a square not exceeding the dimensions of a fresh fuel assembly or to a cross sectional area not exceeding that of a 22.5 cm (8.85 in.) diameter cylinder.

5.2.7.3 Associated with the dismantling operation, one storage position will be permitted for fuel rods or components removed from the assembly provided that:

- The assembly and associated rod storage position shall be separated from each other and from any other fissile material by a minimum of 1 foot surface-to-surface.
- Fissile material and fuel rods or components in the associated storage positions shall be restricted to a square not exceeding the dimensions

of a fresh fuel assembly or to a cross sectional area not exceeding that of a 22.5 cm (8.85 in.) diameter cylinder.

- Only one fuel rod may be inserted or removed from the storage position at a time.
- A maximum of 75 fuel rods shall be permitted in the rod storage position.

5.2.7.4 Fuel assemblies to be studied shall meet the following:

- Each assembly shall be of the enriched PWR type with a 15 x 15, or 17 x 17 square pin lattice not greater than 8.6 inches on a side (further identified as Babcock & Wilcox Mark B or Mark C canless assemblies).
- The maximum initial enrichment in an unirradiated fuel assembly shall not exceed 4.05 wt%.

5.2.7.5 Other PWR or BWR fuel assemblies which do not meet the above listed requirements may be studied, provided:

- The unirradiated, fully reflected fuel assembly (fueled with UO2 only) with all control rods removed is shown by an appropriate Nuclear Criticality Safety evaluation to be subcritical by at least 5% (k-effective < 0.95).
- The fuel assembly is shown by an appropriate safety evaluation to be subcritical by at least 5% k-effective < 0.95) under specific conditions of disassembly.

5.2.7.6 BWR fuel assemblies may be received and studied provided:

- They are evaluated pursuant to the previous section (5.2.7.5), or
- The BWR fuel assemblies have a maximum initial unirradiated enrichment of 4.05 wt% U-235 and have a cross sectional area not exceeding that of a 22.5 cm (8.85 in.) diameter cylinder.

5.2.8 <u>Mixed Uranium and Plutonium Limits at LTC:</u>

Fuel (other than fuel contained in irradiated fuel rods) containing or potentially containing uranium and plutonium will be handled based on units. Each unit will be limited to total fissile material based on the plutonium weight percentage as shown below:

$$\frac{{}^{233}U\ (g) + {}^{235}U\ (g)}{350} + \frac{{}^{239}Pu\ (g) + {}^{241}Pu\ (g)}{220} \leq 1$$

LTC Building B is limited to 40 units, excluding the hot cells, underwater storage, in-ground storage tubes and the examination of commercial fuel assemblies. Each unit shall be separated by a minimum of 8 inches edge-to-edge and 24 inches center-to-center.

5.2.9 LTC Hot Cell Operations

The hot cells shall be limited to one of the following:

5.2.9.1 Three units, as defined in 5.2.8, in Hot Cell No. 1, provided the units are separated by a minimum of 12 inches edge-to-edge and one unit in each of the other hot cells.

5.2.9.2 An irradiated fuel assembly and its associated rod storage positions may be withdrawn from the pool into Hot Cell No. 1 provided that free drainage of water from the assembly and rod storage position and a minimum of 1 foot separation between the assembly and rod storage position and assembly or rod storage position and from other fissile material is assured.

5.2.9.3 Two units in Hot Cell No. 1 may have a total of 64 fuel rods each, stored, provided that rods shall be confined within a cross sectional area not exceeding that of a 22.5 cm (8.85 in.) diameter cylinder, drainage of any free water within the unit shall be assured and the units must be maintained 1 foot from each other and any other SNM in the cell.

5.2.10 Storage Tubes Inside LTC Buildings

SNM in storage tubes shall be limited to the values specified in 5.2.8 for each tube. Storage tubes shall be spaced a minimum of 17 inches center-to-center (except for one pair of tubes which may be spaced a minimum of 16.5 inches), are approximately 5 inches in diameter, and are totally immersed in concrete.

5.2.11 Shipment and Handling of Commercial (PWR and BWR) Fuel Assemblies

After examination, assemblies, partially dismantled assemblies, fuel rods, and scrap generated during destructive examination shall be handled according to the following conditions:

5.2.11.1 Fuel rods, including fuel rod segments may be placed in any available hole in a fuel assembly, including instrumented and control rod guide tube positions, i.e., 225 and 285 fuel rods in Mark B and Mark C assemblies, respectively. Fuel rod segments shall have their ends sealed, and shall be encapsulated in steel tubing with ends sealed, prior to insertion into an available hole in a fuel assembly.

5.2.11.2 Unirradiated assemblies may be reassembled (one rod at a time) for later use.

5.2.11.3 Assemblies, including partially dismantled assemblies, shall be loaded into shipping casks approved for such assemblies for shipment.

5.2.11.4 Scrap, including rod segments, shall be handled according to approved procedures and limits.

5.2.12 Outside Storage at LTC

5.2.12.1 Outside storage consists of underground storage, shipments and the fenced outside storage area located adjacent to Building J.

5.2.12.2 Radioactive material stored in underground storage tubes shall be limited to the unit values specified in Paragraph 5.2.8 per tube. Tubes shall be spaced 20 inches center-to-center and are 5 inches in diameter and totally immersed in concrete.

5.2.12.3 Each shipment of fissile material being stored outside must conform with all license requirements for the type of shipping container. Additionally, each shipment must be nuclearly isolated from all other SNM.

5.2.13 Dry Waste Handling at LTC

Dry Waste is waste that is free of liquids but not necessarily free of hydrogenous material. Dry waste shall be accumulated in steel drums or steel containers with a minimum size of 30 gallons (114 liters) and maximum of 45 grams SNM per container. These containers may be located throughout the laboratories as required to collect contaminated laboratory waste. Filled containers are transferred to Building J. Containers of unirradiated SNM shall be gamma scanned before transfer to verify that the 45 grams SNM per container limit is not violated. Drums containing irradiated SNM are not gamma scanned. The amount of SNM per container is based on a mass balance difference. The amount of waste generated within any hot cell or unit handling irradiated fuel is equal to the difference between the mass of SNM transferred into the unit and the amount transferred out of the unit. Dry waste with less than 0.5 grams of unirradiated SNM per container may be stored in the fenced, locked and paved outside storage area adjacent to Building J.

APPENDIX TO CHAPTER 5

DESIGN CRITERIA FOR NUCLEAR CRITICALITY SAFETY

I. Introduction

This Appendix to Chapter 5, Nuclear Criticality Safety (NCS), describes design criteria used to ensure Nuclear Criticality Safety. After a general section, which describes overall objectives, sections are given to describe means of NCS control, methods of NCS control, and criteria for acceptability of an NCS control.

II. General

The policy for Nuclear Criticality Safety (NCS) design is to: 1) ensure that all risks of a criticality for each operating system have been <u>identified</u>, 2) ensure that any risk which cannot be <u>eliminated</u> is <u>minimized</u> by selecting the highest order Means and Method of criticality control feasible, and 3) ensure that each risk is <u>acceptable</u> by strict adherence to the Double Contingency Principle. Documentation of the preceding for new or modified systems will be provided by the Nuclear Criticality Safety Analyses of the systems. The Integrated Safety Analysis Project (ISAP) will document assurance that existing systems satisfy the Double Contingency Principle and thus have acceptable risk.

The Double Contingency Principle as adopted by NCS is based on a slight variation of that defined in the American National Standard ANSI/ANS-8.1: "Process designs <u>shall</u> incorporate sufficient factors of safety to require at least two unlikely, independent and concurrent changes in process conditions before a criticality accident is possible." There are, however, systems which cannot feasibly utilize classical double contingency protection. Nuclear Criticality Safety for such systems is assured through defense-in-depth to prevent unwanted changes in any one process condition that might adversely affect system safety. Defense-in-depth utilizes two or more reliable barriers or controls to protect against such unwanted changes. Defense-in-depth is enhanced through diversity and redundancy of barriers and controls. The barriers or controls used for defense-in-depth shall be reviewed to ensure that they are not subject to common mode failure (i.e., one malfunction could lead to loss of two barriers or controls). Any deviation from the Double Contingency Principle or of a defense-in-depth protection where Double Contingency is not feasible shall require approval of the Manager of Nuclear Criticality Safety, the Change Review Board, and the U. S. Nuclear Regulatory Commission (via a license amendment).

III. Means of NCS Control

Criticality risks are minimized by selecting the highest order Means and Method of criticality control feasible. The four Means of criticality control, listed in the order of preference, are:

1. Passive Engineered Control (most preferred) uses fixed design features or devices that take advantage of natural forces such as gravity, ambient pressure, etc. No human intervention is required except for maintenance and inspection.

2. Active Engineered Control uses add-on, active hardware, i.e., electrical, mechanical, or hydraulic, to sense parameters and automatically secure the system to a safe condition. No human intervention is required during operation.

3. Enhanced Administrative Control relies on human judgment, training, and personal responsibility for implementation and is augmented by warning devices (visual or audible) which requires human action according to a procedure.

4. Simple Administrative Control (least preferred) relies on human judgment, training, and personal responsibility for implementation each time the control function is needed.

IV. Methods of NCS Control

There are twelve (12) recognized Methods of criticality control; these are also referred to as parameters which may be controlled for Nuclear Criticality Safety purposes (i.e., controlled parameters). When evaluating an SNM bearing system for criticality safety, each of these parameters will be assumed to be at its optimum condition (i.e., most reactive condition) unless specified and acceptable controls are implemented to limit the parameters to certain values. Heterogeneous effects are considered when evaluating a controlled parameter. The 12 Methods are listed below in an approximate order of preference. The order is approximate since the Method of control must be considered in concert with a Means of control (i.e., a passively engineered Method might be preferred to a higher order Method which is maintained by simple administrative Means). The 12 Methods are:

1. **Favorable Geometry Control** is usually the most preferred method and is achieved by increasing neutron leakage by limiting dimensions of a piece of equipment or fuel arrangement. Equipment relying upon favorable geometry for control shall include adequate factors of safety to assure against failure under foreseeable loads and accident conditions. Favorable geometry equipment shall be checked for proper dimensions prior to being released for use; the results shall be documented and maintained on file.

2. **Spacing Control** is almost always needed to specify relative locations; it is a method of limiting interaction between SNM accumulations by separation. Where spacing control is required, a passive engineered devise (e.g., a spacer or bumper) is

the preferred method of control and shall be used where feasible. If not feasible, enhanced administrative control may be utilized and should include such items as procedural instructions, postings, and visual indicators.

3. **Volume Control** is achieved by positive control over the contained volume of SNM to an acceptable value. Equipment relying upon its volume for control shall include adequate factors of safety to assure against loss of that volume under foreseeable loads and accident conditions. The equipment volume shall be checked prior to being released for use; the results shall be documented and maintained on file.

4. **Fixed Neutron Absorber Control** is a method of increasing neutron absorption in material by placing a solid absorber (poison) in the system and includes use of "poison fixtures" as well as taking credit for the neutron absorption properties of structural materials or neutron "poisons" incorporated in product or both. Fixed neutron absorbers, referred to as neutron poisons, may be used following previously described criteria in section 5.2.5.3. These criteria ensure that neutron poisons are adequate, present when needed, and will remain effective over their intended life.

5. **Piece Count Control** is a method of limiting fuel mass and/or geometry by limiting the number of containers or components with known amounts of SNM and/or fixed geometries. Piece count control relies on both manufacturing precision and the accompanying quality control that assures a quality product. Manufacturing variabilities and measurement uncertainties shall be considered when using piece count as a method of control.

6. **Mass Control** is a method of limiting the amount of SNM at a given location to an acceptable value. When the mass of SNM in a unit is utilized for assuring Nuclear Criticality Safety of the unit, the allowed mass shall either meet the safety factors of section 5.2.5.3 of this chapter or the k_{eff} limits of this appendix. Measurements of mass shall satisfy the five requirements for measurement given in Section V of this appendix. SNM mass logs or other methods of showing compliance with mass limits shall be maintained at units under mass control.

7. **Moderation Control** is achieved by limiting or excluding either interstitial (within the SNM) or interspersed (between SNM units) moderating materials or both. Moderating material is primarily hydrogenous substances but also includes such substances as carbon and beryllium. Any use of moderation control shall comply with the criteria and requirements of sections 5.2.5, 5.2.5.1, 5.2.5.2, and 5.2.5.3 of this chapter. In addition, ANSI/ANS-8.22-1997 is committed to with regard to moderation control.

8. **Concentration Control** is achieved by knowing and controlling the SNM concentrations in hydrogenous liquids to an acceptable value. When concentration control is utilized, the concentration shall be determined by sampling and analysis techniques meeting the five requirements of Section V of this appendix or by instrumentation which has been properly maintained and calibrated. Defense-in-

depth shall be utilized to prevent transfers of unacceptable concentrations and to prevent potential unsafe precipitation or concentration. The analysis will consider the solubility limits of the SNM composition and possible concentrating events (e.g., precipitation, evaporation, settling, chemical phase change) and will establish controls necessary to prevent such events as necessary.

9. **Material Specification Control** is a control based on consideration of the physical or chemical composition of material such that the U-235 density and neutron absorption of other materials within the compound are known (e.g., metal versus oxide versus nitrate, etc.). Any use of material specification control relies on both manufacturing precision and the accompanying quality control. Manufacturing variabilities and measurement uncertainties shall be considered when using material specification as a method of control. Possible misidentification shall be considered for feed materials using the feed material specification as control.

10. **Uranium Enrichment Control** utilizes the inherent differences in critical attributes (critical dimensions, mass, etc.) of uranium at different enrichments of U-235. Any NCS control based on knowledge of uranium enrichment shall be made only after appropriate defense-in-depth measures are in place to assure segregation of different enrichments.

11. **Soluble Neutron Absorber Control** is a method of increasing neutron absorption in material by placing a soluble neutron absorber (poison) in a liquid system. Soluble neutron absorbers are only used as secondary NCS control. However, when used, appropriate measurements shall be used to assure their initial presence and their continuous presence at the correct concentration.

12. **Reflector Control** is a method of control which limits neutron return back into an SNM bearing system. It is the least desired Method since all credible reflectors must be considered in each nuclear criticality analysis, section 5.2.4 of Chapter 5, through consideration of type, thickness, and location. All degrees of interspersed moderation, which includes full water reflection, shall always be assumed unless it can be demonstrated that certain degrees of interspersed moderation are not credible. However, some reflectors, such as concrete, can be adequately controlled or partially eliminated from certain areas to render this an acceptable Method. When reflection control is used, the controls to prevent the presence of the potential reflectors are identified as IROFS

In the application of these methods, credit may be taken for certain manufacturing or process parameters as controls (e.g., physical process, chemical properties, etc.). When so utilized, this credit is predicated upon the following requirements:

1. The bounding assumptions are defined and limits established based upon established physical, chemical, or scientific principles and/or facility specific experimental data supported by operational history.

2. Such controls are identified as IROFS, and

3. The process variables are shown in the ISA Summary to be controlled by IROFS.

Each of the above Methods is associated with a Means described in the preceding section. If there is more than one Means possible for a given Method, the highest order Means available and feasible is used. Each control must also be shown to be acceptable as described in the following section.

Each Method, when based on a calculated k_{eff}, has a set of limits, two of which must comply with license limits described in Section 5.2.3 of Chapter 5. The limits are:

1. **Failure Limit** is the critical or just subcritical value. It is determined by an approved technique described in Chapter 5, Section 5.2. Parameters of some systems can be varied through their credible ranges without resulting in a critical system; in such situations there is no Failure Limit.

2. **Safety Limit** is the value of the controlled parameter that will not exceeded unless more than one unlikely, independent and concurrent changes in process condition (contingencies) have occurred. The k-effective values that are associated with this limit are listed in Section 5.2.3.

3. **Limiting Condition of Operation (LCO)** is the value that will not be exceeded unless a contingency has occurred. The LCO is established on a system by system basis. The LCO is based upon the sensitivity of k_{eff} to variations of the controlled parameter (method of control) and upon the ability to detect and control variations to assure that the Safety Limit is not exceeded (see 5.2.3).

4. **Routine Operating Limit (ROL)** is the implementing value that may not be greater than the LCO and should help ensure that a violation of the LCO is unlikely.

V. Acceptability of NCS Controls

The design objective of any Nuclear Criticality Safety control scheme which seeks to minimize criticality risks is to select the highest order Method and Means feasible and then to assure that the controls selected are acceptable by virtue of:

1. Being Functionally Available,

2. Remaining Functionally Available,

3. Having Malfunction Detection and Corrective Systems, and

4. Being Documented.

The preceding four requirements for control acceptability are referred to as "The Four-Way Test."

To be functionally available, an NCS control shall have:

1. Requirements specified in an approved NCS safety analysis,

2. Pre-operational verification that engineered control requirements have been met and are in place,

3. Pre-operational validation of active systems for engineered and enhanced administrative controls that demonstrates that the active systems function as intended,

4. Written criteria of use for any administrative control (e.g., operating procedures and posted NCS signs), and

5. An operational review of changes (can be waived by permission of the NCS Manager) to ensure that NCS requirements are both understood by operations and can be followed without a significant potential of failure.

6. In addition, operators shall have successfully completed training for any administrative controls.

Where a measurement is required as part of the functionality of an NCS control, additional requirements are necessary to assure that the control is functionally available. These additional requirements are:

1. Assurance that any samples drawn are representative of the material being measured,

2. Assurance that measuring systems utilized have the necessary accuracy and precision for the material being measured,

3. Assurance that measuring systems being utilized are properly maintained and that technicians involved with the measurements are appropriately certified/trained, and

4. Assurance that measurement results are accurately reported and accurately received.

5. The use of two independent measurements or the analysis of two independent samples to document compliance shall be performed unless the analysis of failure modes and safety margins specifically justifies fewer.

To remain functionally available, an NCS control shall be designed to:

1. Make operator errors, equipment failures and process malfunctions unlikely,

2. Utilize preventive maintenance, following a schedule and according to approved criteria, of any engineered control,

3. Have periodic testing, following a schedule and according to approved criteria, of the active system of any active engineered or enhanced administrative control, and

4. Have periodic audits and surveillance.

5. In addition, operators shall have periodic retraining where administrative controls are used.

Malfunction Detection and Corrective Systems shall be provided for all control systems. Required elements of the Malfunction Detection and Corrective Systems are:

1. Education/Training Programs for all those involved,

2. Routine Audit and Inspection Programs, and

3. A Cause and Corrective Action System which seeks root causes of any problems affecting NCS and assures effective corrective actions.

Two additional elements that are utilized where feasible are:

4. Controls that are designed to fail to a safe, observable condition, and

5. Use of instruments and alarms together with use of automatic compensatory actions, if possible.

The final element of the Four-Way Test for acceptability of an NCS control is Documentation. Documentation shall be comprehensive, retrievable, and current. The Four-Way Test provides a systematic way to judge the acceptability of NCS controls. As previously stated, risks are first minimized through use of acceptable controls and then made acceptable by demonstration that the system satisfies the Double Contingency Principle.