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August 17, 1984

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Attention: Ms. E. G. Adensam, Chief
Licensing Branch No. 4

Subject: McGuire Nuclear Station
Docket Nos. 50-369 and 50-370
NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants"

Dear Mr. Denton:

On December 22, 1980, Mr. D. G. Eisenhut (NRC/ONRR) issued a letter requesting that Duke Power Company review its controls for the handling of heavy loads to determine the extent to which the guidelines of NUREG-0612 were satisfied at McGuire Nuclear Station, identify the changes and modifications that would be required in order to fully satisfy those guidelines, and provide information documenting the results of our review and implementation of the required changes and modifications. NRC Generic Letter 81-07, "Control of Heavy Loads", was issued on February 3, 1981 correcting several minor errors in the December 22, 1980 letter.

NUREG-0612, Section 5.1, gives recommended guidelines for the control of heavy loads which assure that either (1) the potential for a load drop is extremely small, or (2) for each area addressed certain specified evaluation criteria are satisfied. Toward this end, the NRC developed a defense-in-depth approach for controlling the handling of heavy loads which encompasses an intent to prevent as well as mitigate the consequences of postulated accidential load drops. The subsections provide guidelines on how the defense-in-depth approach may be satisfied for various plant areas. Section 5.1.1 identifies several general guidelines related to the design and operation of overhead load-handling systems in the areas where spent fuel is stored, in the vicinity of the reactor core, and in other areas of the plant where a load drop could result in damage to equipment required for safe shutdown or decay heat removal. Sections 5.1.2 through 5.1.5 provide specific guidelines concerning the design and operation of load-handling systems in the spent fuel pool area-PWR (Section 5.1.2), containment building-PWR (Section 5.1.3), reactor building-BWR (Section 5.1.4), and other areas containing safe shutdown equipment (Section 5.1.5). In addition, in order to assure safe handling of heavy loads in the interim period until measures at operating plants are upgraded to satisfy the guidelines of Section 5.1, Section 5.3 gives various interim protection measures to be implemented.

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Duke Power Company has submitted responses to the NRC's December 22, 1980 letter via Mr. W. O. Parker Jr.'s letters to Mr. H. R. Denton (NRC/ONRR) dated June 2, 1981; August 5, 1981; October 8, 1981; November 23, 1981; January 15, 1982; March 3, 1982; June 4, 1982; and July 26, 1982. Ms. E. G. Adensam's letter of September 9, 1982 transmitted a draft technical evaluation report (dated August 31, 1982) which was developed by the Franklin Research Center (under a technical assistance contract to the NRC) to assess McGuire's conformance to the general load-handling guidelines of NUREG-0612 Section 5.1.1 (corresponding to Section 2.1 of enclosure 3 of the December 22, 1980 letter), and to the interim protection measures of NUREG-0612 Section 5.3 (corresponding to enclosure 2 of the December 22, 1980 letter). My letter of November 1, 1982 provided a response to this draft TER, as well as submitting further informaton requested by the December 22, 1980 letter. As a result of continued evaluation of McGuire Nuclear Station a revised dradt TER (dated January 12, 1983) was transmitted by Ms. Adensam's June 10, 1983 letter. This draft TER found that load-handling operations at McGuire can be expected to be conducted in a reliable manner generally consistent with the staff's objectives as expressed in the general load-handling guidelines, and that (with certain exceptions) the interim protection measures have been satisfactory implemented. However, the draft TER identified several unresolved items for which additional information was requested. My letter of July 15, 1983 supplied a partial response to these items; additional information is provided below.

Guideline No. 1 - Safe Load Paths (NUREG-0612, Section 5.1.1(1))

Safe load paths for all major equipment handled in the reactor building (including the reactor building polar cranes) were developed and were in use in April, 1983. Load paths for the ice condenser bridge cranes were developed and in use in April, 1983. However, it has subsequently been determined (as shown in attached FSAR Figures 1.2.2-14, 1.2.2-15, and 6.2.2-1) that the ice condenser bridge cranes do not handle heavy loads in the areas where spent fuel is stored, in the vicinity of the reactor core, or in other areas of the plant where a load drop could result in damage to equipment required for safe shutdown or decay heat removal (load drop would be onto a three foot thick concrete floor (bottom deck) and could only damage ice condenser equipment), and therefore it is concluded that the requirements of NUREG-0612 are not applicable. My July 15, 1983 response indicated that visual aids would be provided to assist the crane operator in ensuring that the designated safe load paths developed in the auxiliary building which were not painted on the floor (i.e. for fuel handling area cranes, A108A and A111A) are actually followed. Procedures to implement the use of these visual aids were to be in effect by September 1, 1983. However, lines designating the safe load paths for the fuel handling area cranes have since been painted on the floor (work was completed in March 1984). These lines are only partially covered by plastic during

fuel cask movement. The lines, along with having the designated travel path shown in an enclosure to the procedure and having a handling supervisor verify the load path, are adequate to meet the intent of guideline No. 1. All designated load paths which have been developed in other buildings, including the reactor building polar cranes (and ice condenser bridge cranes although not required as indicated above), are either marked on the floor, roped off, designated in procedures, or are clearly marked on drawings in the lift supervisors handbook. With the submittal of this information full compliance with guideline No. 1 (and consequently Interim Protection Measure No. 2 of NUREG-0612 Section 5.3) has been demonstrated.

Guideline No. 2 - Load Handling Procedures (NUREG-0612, Section 5.1.1(2))
Procedures for handling heavy loads and using safe load paths for cranes located in the reactor building which comply with the requirements of guideline No. 2 were developed and approved/implemented April 29, 1983. With the submittal of this information full compliance with guideline No.2 (and consequently Interim Protection Measure No. 3 of NUREG-0612 Section 5.3) has been demonstrated. Since compliance with the remaining Interim Protection Measures (Nos. 1, 4, 5 and 6) has previously been acknowledged in the January 12, 1983 TER, McGuire has fully implemented all Interim Protection Measures of NUREG-0612 Section 5.3. (Note that although compliance with Interim Protection Measure No. 1 of NUREG-0612 Section 5.3 was not requested by the December 22, 1980 letter, Duke Power Company demonstrated compliance with this measure in response to the August 31, 1982 TER).

Guideline No. 4 - Special Lifting Devices (NUREG-0612, Section 5.1.1(4))
As indicated in my July 15, 1983 response, the following are the special lifting devices at McGuire which must be evaluated for compliance with the requirements of ANSI N14.6-1978 (As supplemented by NUREG-0612, Section 5.1. 1 (4)):

a. Reactor vessel head lifting rig and load cell
b. Reactor internals lifting rig
c. Reactor Coolant Pump motor lifting rig
d. Control Rod Drive Mechanism (CRDM) missile shield lifting rig

The August 31, 1982 draft TER supplied a list of the specific sections of ANSI N14.6-1978 which must be addressed to determine compliance with guideline No. 4. Attachment No. 1 provides an assessment of each of these sections for items c and d above which were designed and constructed by Duke Power Company. Items a and b were manufactured by Westinghouse, and Duke Power is currently pursuing additional information from them necessary for the evaluation. Once this information is received an assessment will be made and submitted to the NRC. NRC/OIE Information Notice 83-71,

"Defects in load-bearing welds on lifting devices for vessel head and internals", was reviewed with respect to the McGuire reactor vessel head and internals lifting rigs. An inspection procedure meeting the requirements of ANSI N14.6-1978 has been written for these lifting devices. The lifting devices are visually inspected annually. All load bearing parts and welds are inspected and any part or weld that appears suspicious or for which a defect is detected is subjected to an appropriate non-destructive examination by QA personnel. Duke Power Company believes the preventative maintenance program in place at McGuire is adequate to insure the structural integrity of the reactor head and internals lifting devices. Further, these devices were load tested by Westinghouse to 125% of maximum load (NUREG-0612 requires that the lifting device be tested in accordance with ANSI N14.6, i.e. a load test of 150% of maximum load prior to initial use). In addition, the questions raised in the information notice will be addressed in the above mentioned NUREG-0612 lifting rig analysis for these rigs which has been requested from Westinghouse. Since compliance with the remaining general load-handling guidelines (Nos. 3,5,6, and 7) has previously been acknowledged in the January 12, 1983 TER, McGuire has demonstrated full compliance with all guidelines of NUREG-0612 Section 5.1.1 (except as noted for guideline No. 4).

Duke Power Company's July 26, 1982 submittal contained information in accordance with the specific requirements of Section 2.3 of enclosure 3 of the December 22, 1980 letter (corresponding to NUREG-0612 Section 5.1.3) and indicated that further information concerning reactor vessel head drop analyses would be forwarded later. Based on the evaluation of Safe Shutdown and Decay Heat Removal equipment, it was determined that the following approach would satisfy the intent of NUREG-0612. The polar crane is assumed to be capable of a drop at any point within the crane wall. Since the head is the largest item that can be dropped, all smaller loads would be covered by the reactor vessel head. This simplifies the analysis since we assume any/all equipment can be damaged by a load drop. The July 26, 1982 submittal stated that only two of the crane/load combinations possible were to be considered because they envelope the others: (1) dropping of the reactor vessel head onto the vessel flange, and (2) oblique drop of the reactor vessel head onto the upper internals. These analyses were performed by Westinghouse with the following results. Duke Power supplied Westinghouse with drawings of the interior concrete layout and also provided information concerning the height of drop and medium through which the head would fall, and a sketch showing the relative elevations of structures and components in the safe load path of the vessel head from the reactor vessel to the head storage stand. The information available within Westinghouse included: the masses of the objects involved in the impact, the stiffnesses of the reactor vessels, nozzles, supports, and loop piping, the length of the reactor vessel guide studs; and the details of the vessel heads, reactor vessels, and vessel nozzles. The information was used to develop the conditions and scenarios for which the postulated drop accident was evaluated. Attachment 2A is the report documenting this analysis. The

results of the analysis shows that the reactor vessel nozzle stresses caused by the head drop are within allowable limits. The reactor vessel support impact loads, however, exceed the faulted condition vertical allowable loads. Consequently, the reactor vessel supports had to be evaluated to determine the effects of these high loads on the ability of the system to maintain adequate residual heat removal and, thereby, prevent excessive radiation releases. Maintenance of core cooling capability in this case is dependent upon the loop piping and essential auxiliary piping to remain intact following the vertical displacement of the reactor vessel. The purpose of this support/piping investigation was to determine the maximum vessel displacement and the subsequent effects on the essential piping. This additional analysis was performed by Duke Power and demonstrated that primary loop piping, auxiliary piping and vessel supports are adequate to maintain core cooling capabilities. A third load drop of the reactor vessel head was considered. This case is a drop on the operating floor which causes concrete spalling and subsequent possible damage to equipment on lower levels of the Reactor Building. This case was not mentioned in earlier submittals covering the Reactor Building. Attachment 2B provides a summary of this case's analysis which concludes that the postulated reactor vessel head drop does not penetrate the operating floor; some scabbing does occur on the underside, however, the structural stability and functional requirements are maintained. Additionally, attachment 2C provides summary information on analyses performed by Duke Power Company for reactor vessel head drop on vessel internals primary shield wall, load swing into refueling canal wall, and drop onto the reactor internals. Attachment 2D provides the radiological dose consequence for a reactor vessel head drop, and attachment 2E provides a criticality analysis.

One means of complying with the guidelines of NUREG-0612 sections 5.1.2, 5.1.3, 5.1.4, or 5.1.5 is upgrading of the crane and lifting devices to conform to the single-failure-proof guidelines of section 5.1.6, one of which was NUREG-0554, "single-failure-proof cranes for nuclear power plants". Mr. D. G. Eisenhut's letter dated December 19, 1983 (NRC/ONRR Generic letter 83-42, "Clarification to generic letter 81-07 regarding response to NUREG-0612") indicated that in the course of reviewing crane designs against NUREG-0554, concerns of a generic nature were identified which indicate that NUREG-0554, until revised, may be deficient in assuring single failure proof cranes. It was stated that this aspect of single failure proof cranes would be part of the NRC's review of Duke's submittals if we take credit for a single failure proof crane to satisfy NUREG-0612. In regard to this, McGuire has not and will not use single failure proof cranes as compliance to NUREG-0612.

Should there be any questions in this matter, please advise.

Very truly yours,

Hal B. Tucker

Hal B. Tucker

PBN/mjf

Attachments

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ATTACHMENT 1

ASSESSMENT OF SPECIAL LIFTING DEVICES FOR COMPLIANCE WITH GUIDELINE NO. 4

The following is an assessment of the Reactor Coolant Pump Motor Lifting Rig and the Control Rod Drive Mechanism (CRDM) Missile Shield Lifting Rig for Compliance with the requirements of ANSI N14.6-1978 (as supplemented by NUREG-0612, Section 5.1.1 (4)). Strict interpretation of compliance of existing special lifting device design with the criteria of ANSI N14.6-1978 cannot be made. Accordingly, only those sections directly related to load-handling reliability of the lifting devices need be addressed. Several sections of ANSI N14.6-1978 do not contain requirements concerning load-handling reliability: Scope (Section 1), Definitions (2), Design Considerations to Minimize Decontamination Efforts (3.4), Coatings (3.5), Lubrication (3.6), Inspector's Responsibilities (4.2), and Fabrication Considerations (4.3). Evaluation of compliance with Section 6 (Special Lifting Devices for Critical Loads) need not be included since no load has been determined to be a "critical load." The specific sections of ANSI N14.6-1978 referenced below are those stated in the August 31, 1982 TER for which compliance or equivalence must be demonstrated in order to determine compliance with guideline no. 4. (Note that the TER referenced the applicable ANSI sections both by number and brief description. However, several of the numbers did not correspond to the sections indicated by the description. For these cases Duke Power assumed the descriptions referenced the section for which compliance or equivalence with was intended).

Section 3.1.1: Design calculations and drawings (and notes on drawings) cover specification criteria defined in Section 3.1.1.

Section 3.1.2: The design specifications, calculations, and drawings identify all load bearing (i.e. critical) components, define their critical characteristics, and specify material identification, qualification, and control for these components. There are general statements about material requirements for other (non-critical) components. Fabrication practices are discussed in the specification and on the drawings (i.e. notes on drawings). In-process testing and inspection with acceptance criteria is also specified, as well as requirements for final product testing and inspection to applicable acceptance criteria. Duke's QA program was followed on the design of the special lifting devices and is in accordance with 10CFR 50, Appendix B.

Section 3.1.3: Signed stress analysis exist which demonstrates the adequacy of the special lifting devices and their components with respect to any loads that may be imposed upon them during the performance of their functions. These analysis also demonstrate appropriate margins of safety.

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Section 3.1.4: All special lifting devices are unique to the item they were designed to lift. Any repairs made would also be unique to each lifting device; therefore, all repairs would be handled on an individual basis and would depend on the severity of the problem. Any major repair would be subject to 125% load test.

Section 3.2.1: Duke Power performed an analysis of the McGuire Reactor Coolant Pump Motor Lifting Rig and the CRDM Missile Shield Lifting Rig per ANSI N14.6-1978 Section 3.2.1 as modified by NUREG-0612 Section 5.1.1(4). Both of the rigs are constructed of A-36 steel having a minimum yield strength of 36 KSI and a minimum ultimate strength of 58 KSI. The rigs are required to meet stress design factors of 3 for minimum yield strength and 5 for ultimate strength. Since the ratios of 58 KSI/36KSI and 5/3 are approximately equal (1.61 and 1.67 respectively), checking one of the stress design factors essentially checks the other stress design factor (if one fails both will fail, if one passes both will pass). Therefore, the analysis of each rig consisted of checking the following case: The rig must be capable of lifting 300% (stress design factor of 3) of the maximum static plus dynamic (15% of static) load without exceeding the yield strength of the materials of construction. For the McGuire RCP Motor Lifting Rig (static load = 88.3 kips, static + dynamic load = 102 kips) analysis determined that the rig would be able to meet this case (300% = 306 kips), and therefore is also capable of lifting 500% (stress design factor of 5) of this load without exceeding the ultimate strength of the materials. For the McGuire CRDM Missile Shield Lifting Rig (static load = 145 kips, static + dynamic load = 167 kips) analysis determined that the rig would not be able to meet this case. The rig was determined to be capable of lifting 120% (201 kips) the maximum static plus dynamic load or 139% (201 kips) the maximum static load without exceeding the yield strength of the materials of construction. Consequently, the rig would also not be capable of meeting the ultimate strength stress design factor. The McGuire CRDM Missile Shield Lifting Rig (and RCP Motor Lifting Rig) was designed and built prior to the publication of ANSI N14.6-1978, and therefore is not designed in accordance with that standard. In addition, these rigs do not carry shipping containers of nuclear material. However, the NRC has taken the position (Ref. "Synopsis of issues associated with NUREG-0612", dated May 4, 1983 which was transmitted with the January 12, 1983 TER) that for special lifting devices subject to NUREG-0612, it should be able to be demonstrated that, from a design standpoint, they are as reliable as a device for which ANSI N14.6 was developed. Although not originally specified to be designed in accordance with ANSI N14.6, the special lifting devices were provided by Duke Power Company in accordance with appropriate quality assurance and quality control procedures, for a specific application. These lifting rigs meet their applicable design

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specifications and therefore it is Duke Power Company's opinion that these rigs are designed to an acceptable factor of safety. This position is consistent with exception No. 1 to guideline No. 4 as given in the above referenced synopsis of issues.

Section 3.2.4: Not applicable to the Reactor Coolant Pump Motor Lifting Rig and the CRDM Missile Shield Lifting Rig because these rigs do not have any load bearing pins, extension links, or adapters. However, all members are designed to meet section 3.2.1.

Section 3.2.5: All slings at McGuire Nuclear Station are inspected and tagged with a color coded I.D. tag annually. The slings comply with ANSI B30.9 (1971).

Section 3.2.6: Temperatures inside the Reactor building will be maintained at all times at a level that will ensure ductile conditions (i.e. brittle fractures won't occur) for the material used in these applications (i.e. carbon steel) in accordance with paragraph AM 218 of the ASME boiler and pressure vessel code, Section VIII Division 2, and therefore impact testing is not required. All temperature requirements of the design specification are met.

Section 3.3.1: Not applicable to the Reactor Coolant pump motor lifting rig and the CRDM Missile Shield Lifting Rig because the lifting rigs are protected from the environment and from galling. Lamellar tearing is not a problem although it is considered in the design process.

Section 3.3.4: The rig's design assures even distribution of load.

Section 3.3.5: All load-carrying components have been fitted with cotter pins or lock pins and/or lockwire.

Section 3.3.6: The spent fuel cask lifting yoke is the only lifting device with a remote actuating mechanism. The yoke is in full view of the operator at all times and, therefore, does not require a position indicator.

Section 4.1.3: Design drawings indicate materials to be used, and the rigs were fabricated to those requirements.

Section 4.1.4: The rigs were fabricated in accordance with design requirements and using generally accepted good practices.

Section 4.1.5: All welders in the Construction Department at McGuire were qualified in accordance with ASME Section IX and their qualifications are documented and controlled in accordance with Quality Assurance procedure (QAP) L-100, entitled the "Welding Program", and QAP I-1, entitled "Qualification of Welders and Operators". Likewise, all procedures are qualified in accordance with the ASME Boiler and Pressure Vessel Code as directed by QAP L-100.

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Section 4.1.6: As directed by the Code of Federal Regulations, Duke Power Company has operated the Construction Department Nuclear Station sites in accordance with a strict Quality Assurance program which meets all the requirements of this section. Quality Assurance procedure F-1 dictates the control of field Construction Procedures (CP). These lifting rigs were fabricated and welded in accordance with CP 57, "Welding of Structural Steel Miscellaneous Steel and Other Steel Construction for Nuclear Structures and Certain Non-Nuclear Structures" and CP 859, "Structural Steel Fabrication, Erection and Inspection".

Section 4.1.7: All structural steel for McGuire were ordered, received, controlled and issued in accordance with the Quality Assurance procedures for Nuclear Safety Related materials. The specific materials for these lifting rigs is designated to be A-36. This material was color coded in accordance with CP 395, "Identification Marking and Issuing of Miscellaneous Steel Stock For QA Condition Uses" and was issued accordingly, although these lifting rigs were not Nuclear Safety Related.

Section 4.1.9: Since these lifting items were not designated Nuclear Safety Related, they did not receive the same inspections that a QA item would have. However, all welds were visually inspected by a qualified welding inspector in accordance with CP 57. This was and is a practice for all welded joints. Section 4.1.7 addresses the adequacy of the materials.

Section 5.1.3: All special lifting devices are covered under our preventative maintenance (PM) inspection program. They are inspected prior to use at the beginning of an outage.

Section 5.1.4: Procedures for the removal or movement of equipment associated with special lifting devices have been developed or are incorporated in the procedures for work on that equipment. The operating procedure does not outline maintenance of the devices, however there is a yearly preventative maintenance which covers inspection. Each lifting device is designed to lift only one piece of equipment and therefore are limited to those applications - procedures specify use of appropriate rigs.

Section 5.1.5: The special lifting devices used at McGuire Nuclear Station are not marked with either the rated capacity or identification numbers; however, each device was designed and built to move only that equipment it is associated with (i.e. one specific use). McGuire does not have any lifting devices which have subparts or subassemblies that can be exchanged or replaced (each unit is all one piece).

Section 5.1.6: Records for each lifting device are maintained on a yearly PM Work Request.

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Section 5.1.7: All special lifting devices are inspected on an as required basis. If any indications are found in the inspection areas, the device will be tagged and removed from service until the problem is resolved.

Section 5.2.1: Duke Power performed an analysis of the McGuire Reactor Coolant Pump Motor Lifting Rig and the CRDM Missile Shield Lifting Rig per the guidelines set forth in NUREG-0612. The analysis of each rig consisted of checking the following case: The rig must be capable of lifting 150% of the maximum static load to which the rig is to be subjected.For the McGuire RCP motor lifting rig (static load = 88.3 kips) analysis determined that the rig would be able to meet this case (150% = 132.5 kips). For the McGuire CRDM Missile Shield Lifting Rig (static load = 145 kips) analysis determined that the rig would not be able to meet this case. The rig was determined to be capable of lifting 125% (181 kips) the maximum static load to which the rig is to be subjected. Based on the above analysis, load tests for the McGuire Reactor Coolant Pump Motor Lifting Rig and the CRDM Missile Shield Lifting Rigs were conducted at Catawba Nuclear Station in March 1984. Non-destructive testing of each rig was conducted at McGuire by the QA Staff. The McGuire Unit 1 and 2 CRDM Missile Shield Lifting Rigs were load tested to 125% (181 kips) maximum static load, and the McGuire Reactor Coolant Pump Motor Lifting Rig (1 rig for the station) was load tested to 150% (132.5 kips), as explained above. After successfully sustaining the load for a period of 10 minutes (no deformation of rig observed), each rig was then subjected to non-destructive testing. No indications were present except for a surface lamination on one of the Unit 1 CRDM Missile Shield Lifting Rig's bolts used to attach the missile shield to the lifting rig. This lamination did not occur as a result of the load test, but rather appears to have been made at the time of manufacture. Even though the bolt successfully carried the test load, it will be replaced. An NDE will be performed on the new bolt prior to placing it in service. The Unit 2 CRDM bolts were found acceptable through NDE testing. In addition, when the RCP motor lifting rig was tested, the turnbuckles used with the rig were mistakenly left off. The turnbuckles will be load tested separately and then inspected. The RCP motor rig can be used without the turnbuckles until they are satisfactorily tested. Results of these tests will be forwarded upon completion. After review of these results we have found the lifting rigs to be acceptable. These lifting rigs meet their applicable design specifications and are designed to an acceptable factor of safety. Although the CRDM Missile Shield Lifting Rigs were not subjected to 150% overload tests (150% = 217.5 kips) for reasons outlined above, the NRC has recognized that the specification of a 150% overload test is somewhat arbitrary and has provided for exceptions to verbatum compliance with NUREG-0612 guidelines via the "Synopsis of issues associated with NUREG-0612" (dated May 4, 1983) which was transmitted with the January 12, 1983 TER. NUREG-0612 Section 5.1.1 (4) also indicates that certain load tests may be accepted in lieu of certain material requirements in

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the ANSI standard. The rigs were tested to 125% overload which has been standard industrial practice for some time.

Section 5.2.2: At present, no spare parts are stocked for special lifting devices. Appropriate measures (i.e. compliance with Section 5.2.2) will be taken should any spare parts be stocked in the future.

Section 5.3.1: Annual load testing per Section 5.3.1 Part (1) for McGuire's lifting devices are omitted in accordance with Section 5.3.1 part (2) of ANSI N14.6-1978. Dimensional testing, visual inspection, and nondestructive testing of major load-carrying welds and critical areas is performed in accordance with Section 5.5 as permitted by Section 5.3.1 part (2).

Section 5.3.2: McGuire intends to load test all special lifting devices after major modications or repairs in accordance with Section 5.3.2 of ANSI N14.6-1978.

Section 5.3.3: McGuire's special lifting devices are designed to specific pieces of equipment and should never be subjected to stress substantially greater than they were designed for. If they were subjected to an over-stressed condition, appropriate measures would be taken.

Section 5.3.6: All special lifting devices are visually inspected by personnel using the device prior to each use as specified in the Lift Supervisor's Handbook.

Section 5.3.7: Maintenance personnel inspect each lifting rig in accordance with PM requirements prior to outages (unless rig was inspected and used within the last 30 days). These rigs are inspected prior to outages (as opposed to every 3 months) in view of ALARA considerations, and since the rigs are only used during outages there is no reason to inspect them at 3 month intervals during power operation in which they wouldn't be used.

ATTACHMENT 2A

DUKE POWER COMPANY
McGUIRE AND CATAWBA NUCLEAR POWER STATIONS
REACTOR VESSEL HEAD DROP ANALYSIS

1.0 SUMMARY

An evaluation of the effects of a postulated reactor vessel head drop accident as described in NUREG-0612 "Control of Heavy Loads at Nuclear Power Plants" was performed for the Duke Power Company McGuire and Catawba Nuclear Power Stations. The accident is postulated to occur during refueling when the head is manipulated above the reactor vessel. The polar crane is postulated to fail and the head is assumed to fall concentrically onto the reactor vessel.

The analysis results indicated that although the reactor vessel primary nozzles would not be stressed above allowable limits, the vessel supports would experience an impact load in excess of the faulted condition allowable load. The individual reactor vessel support loads are tabulated below.

	IMPACT LOAD (KIPS PER SUPPORT)	ALLOWABLE LOAD (KIPS PER SUPPORT)
McGuire 1	13,950	2600
McGuire 2	13,180	2600
Catawba 1	17,190	6600
Catawba 2	18,570	6600

The reactor vessel supports require an evaluation to determine the effects of these high loads on the ability of the system to maintain adequate core cooling capability. Westinghouse recommends that the reactor vessel supports be evaluated using an impact load generated by an equivalent system time history method. For this method the energy loss at impact is considered by the reduction of velocity of the system at impact based on conservation of momentum. The response of the reactor vessel supports would be represented by a bilinear load-displacement curve. Additional assumptions and procedures specific to the McGuire and Catawba Plants would be decided upon later.

2.0 METHODS

The following assumptions are applicable to the dropped head accident analysis performed for the McGuire/Catawba Plants.

1. A concentric drop of the reactor vessel head onto the reactor vessel is assumed. The high stiffness of the vessel mating flange and the vessel cylinder between the nozzles and the mating flange would cause any impact load to be evenly distributed to the four vessel supports. Since in a non-concentric drop some of the impact would be taken by the refueling cavity concrete floor, the concentric drop configuration is the upper bound for the loading of the reactor vessel supports.

2. The buckling of the reactor vessel head guide studs is assumed to have an insignificant effect on the vessel head as it drops. The guide studs have been shown in WCAP 9198 "Reactor Vessel Head Drop Analysis" not to cause the head to rotate a significant amount when they do not engage the head properly. Similarly, the buckling of the CRDM drive rods has been shown not to impart to the fuel assemblies a load that could cause damage to the integrity of the fuel rods.

3. The vessel head is assumed to be rigid and deflection of the head at impact is neglected.

4. When dropping through water, the drag coefficient of the vessel head is that of a hollow half-hemisphere. Additionally, only one-half of the buoyant force is taken into account when the head falls through water. This is because the head is out of the water at the start of the drop and the buoyant force is not fully developed until the head is completely submerged.

5. During impact of the vessel head with the reactor vessel, some of the kinetic energy of the head is dissipated. The remainder is assumed to act through the vessel and nozzles to produce the support loads caused by the impact. The dissipated energy can be calculated by equating the momentum of the system before and after impact or the energy loss can be taken into account by applying to the kinetic energy available just before impact an energy dissipation factor.

   The reactor vessel nozzles were evaluated using the impact load calculated by the method given in WCAP 9198. This WCAP gives a method for taking into consideration the energy dissipated at impact. The equation for the energy dissipation factor used in WCAP 9198 includes the mass of the vessel head and the mass of the vessel shell from the mating flange to the underside of the primary nozzles plus the mass of the nozzles. No credit is taken for the remainder of the reactor vessel, the internals, fuel, or water in the vessel. The energy dissipation factor used in the McGuire/Catawba analysis was expanded to include the available mass of the other parts of the vessel, fuel, water, and internals.

6. The scenarios for considering the maximum effects of a dropped head are as follows:

   a. The polar crane is assumed to fail when the head is at the highest point over the reactor vessel prior to lateral motion. This drop case was specified by Duke Power in letter MCN-83M-25 (CN-83M-34) as being 18 feet through air followed by 24 feet through water.

   b. The polar crane is assumed to fail after the vessel head engages the guide studs and the refueling canal is drained. The vessel head will fall in this case 16 feet through air.

The velocity at impact of 'a' above was determined to be equivalent to that for a drop entirely through air from a height of 15.7 feet for McGuire 1 and Catawba 2 and 14.7 feet for McGuire 2 and Catawba 1. Therefore, the drop through air of 16 feet was used as the worst case drop scenario for the McGuire and Catawba dropped head accident.

ATTACHMENT 2B

DUKE POWER COMPANY
MCGUIRE AND CATAWBA NUCLEAR POWER STATIONS
REACTOR VESSEL HEAD DROP ON THE OPERATING FLOOR

DATA SUMMARY

Initial Conditions/Assumptions

a. Weight of heavy load: 289,772$^{lb.}$; RV Head, Platform and Accessories, Lifting Rig, Load Cell, Crane Hook, Block and Cable

b. Impact area of load:

   1. Flat drop on the operating floor. Impact area of 8603 in^2

c. Drop height: 5.583$^{ft.}$ - height of storage stand plus 6 inches

d. Drop locations: Operating floor

e. Credit for the action of impact limiters: No credit

f. Thickness of floor slab: 2 feet - 6 inches

g. Drag forces: No drag forces are assumed

h. Load combinations: Impact load plus dead weight of the slab

i. Material properties: Main Reinforcing: No. 11 bars, grade 40, Concrete - f'_c = 5000 psi

Method of Analysis:

The operating floor was modeled using a STRUDL space frame finite element model subject to the RV head impact load. The impact load, slab ductility and penetrations were determined based on methods described in Chapter 6 of ASCE manual number 58, "Structural Analysis and Design of Nuclear Plant Facilities" and Williamson and Alvy, "Impact Effect of Fragments Striking Structural Elements". The resulting shears and moments were evaluated using conventional design methods

Conclusion:

The postulated reactor vessel head drop does not penetrate the operating floor, some scabbing does occur on the underside, however, the structural stability and functional requirements are maintained.

ATTACHMENT 2C
DUKE POWER COMPANY
McGUIRE AND CATAWBA NUCLEAR POWER STATIONS
RESULTS OF REACTOR VESSEL HEAD DROP ANALYSIS

DATA SUMMARY

Drop onto canal floor (top of primary shield wall, elev. 747'-4")	No damage to RC System piping. Drop consequences are bounded by Oconee calculations. ISAS Evaluation No. 82-09
Load swing into canal walls	Consequences are assumed to be less severe than drop onto the floor or onto the internals; therefore, no analyses have been conducted.
Drop onto the reactor internals	In reference to Oconee calculations, ISAS Evaluation 82-09, and realizing impact loads to be similar, 100% of fuel core damage would result. Radiological dose calculations were performed and are given in Attachment 2D.

ASSUMPTIONS USED IN McGUIRE/CATAWBA REACTOR VESSEL HEAD DROP ANALYSIS

DATA SUMMARY

a. Weight of Heavy Load: 290,000#; RV Head, CRD Structure, and Platform, Lifting Rig, and Crane Hook and Cable.

b. Impact Areas: Areas were judged similar to those used in the Oconee calculations. Actual calculations for McGuire/Catawba were not necessary.

c. Drop Height: 32 feet - Operating floor to canal floor plus 6 inches of clearance.

d. Drop Locations: Canal floor, RV internals, load swing into the canal walls.

e. Credit for action of impact limiters: No credit. No water is in the canal and the steel liner is ignored.

f. Thickness of walls or slabs: Primary wall, 8 feet; Refueling canal wall, 4 feet.

g. Drag Forces: No drag forces are assumed.

h. Load Combinations: Dynamic Impact; Drop is onto the top of a wall, therefore, slab dead weight is not applicable.

i. Material Properties: 60 ksi reinforcing steel, 5 ksi concrete.

Method of Analysis

No calculational analyses were performed for McGuire or Catawba; however, a review of loads and drop heights was performed, and it was determined that for this portion of the analysis, the consequences of a head drop at McGuire or Catawba were bounded by calculations provided for Oconee.

Conclusions

In the unlikely event of a RV head drop from above the refueling canal, extensive damage to the canal floor (top of primary shield wall) or RV internals would result depending on the location of the drop. Damage to the wall, however, would be localized concrete crushing and would not impact the reactor coolant system piping. Impact of the RV internals would result in extensive damage to the fuel, thus leading to a possible 100% gap activity release. The resultant offsite release has been addressed in Attachment 2D, and procedural changes are being pursued which would require automatic containment isolation in the event of a fuel handling accident of this nature.

Load swing with subsequent impact on the canal walls was considered insignificant in comparison to the previously cited accident scenarios; therefore, no analysis was performed.

In conclusion, under any of the postulated head drop scenarios, the procedural changes referenced earlier will minimize any possible radiological releases. In the event that any RC piping is damaged, redundant RHR train capabilities would maintain water level above the core to provide adequate cooling.

ATTACHMENT 2D
DUKE POWER COMPANY
McGUIRE NUCLEAR STATION

Analysis of Radiological Releases for NUREG-0612 Concerns

1. Initial Conditions/Assumptions

   a. Time of accident following shutdown = 100 hours

   b. Number of fuel assemblies damaged = 193 of 193

   c. Power level prior to shutdown = 3411 (Mwt)

   d. Fraction of Activity in the gas gap = 10% for Iodines and Noble Gases except Kr-85
      30% for Kr-85

   e. Pool water DF = 1 for Noble Gases
      133 for Inorganic Iodine
      1 for Organic Iodine

   f. Iodine gas is made up of: 99.75% Inorganic
      .25% Organic

   g. For discharges released through charcoal filters, filter efficiencies are: 90% for Inorganic
      70% for Organic

   h. The worst 5 percentile meterological conditions are used for dispersion.

2. Methods of Analysis

   a. Releases are calculated from total fission product inventories at the end of core life. Decay of 100 hours is assumed. Decontamination factors, filter efficiencies, dose calculations, etc., are according to Reg. Guide 1.25 methodology.

   b. For release through large open penetrations, all activity released through pool water is assumed to escape from containment.

   c. For releases through the purge system, no isolation was considered. Also, credit was taken for the removal of Iodine by the charcoal beds.

3. Conclusion

   Postulated doses were found to exceed one-fourth of 10 CFR Part 100 limits for all cases. The results of the dose analysis are presented in Table 1.

TABLE 1

Results of Reactor Vessel Head Drop Dose Analysis

For

McGuire

		Site Boundary Doses
1.	Release through Charcoal Filters	Whole Body = 123 Rem Thyroid = 4,590 Rem
2.	Release without Charcoal Filters	Whole Body = 123 Rem Thyroid = 30,524 Rem
3.	Regulatory Limits (1/4 of 10CFR100)	Whole Body = 6.25 Rem Thyroid = 75 Rem

ATTACHMENT 2E
DUKE POWER COMPANY
McGUIRE NUCLEAR STATION

CRITICALITY OF DAMAGED FUEL EVALUATION

I. APPLICABILITY

Oconee, McGuire, Catawba, Cherokee, All Units

II. SCOPE

The purpose of this evaluation is to analyze generically the effect on criticality of physical damage to a fuel array. This applies to both fuel in spent fuel storage pools and fuel in the reactor core.

III. BACKGROUND

One of the basic properties of a finite system in which neutrons are being produced by fission is the effective multiplication factor, K_{eff}. The requirement for criticality in such a system is $K_{eff} = 1$ which makes possible a steady-state fission chain. If conditions are such that more neutrons are lost in each generation than are produced by fission, the system is considered subcritical or $K_{eff} < 1$ and the chain decreases. If more neutrons are produced than are lost in each generation, the system is considered supercritical or $K_{eff} > 1$ and the fission chain increases.

To determine the value of K_{eff}, it is necessary to consider two areas. The first involves K_{∞}, the infinite multiplication factor, which is the ratio of the number of neutrons resulting from fission in each generation to the number absorbed in the preceding generation in a system of infinite size. It is a function of the materials of the system, such as fuel, moderator, coolant, structure, etc., and in a heterogeneous (or lattice) system, the physical arrangement of the materials is also important. The second involves leakage of neutrons out of the system. The larger the system, the less leakage is likely.

The minimum quantity of material that is capable of sustaining a fission chain once it has been initiated by an external source of neutrons, is called the critical mass. The critical mass of material required for a system depends upon a wide variety of conditions, two important items being the amount of fissile uranium-235 present (the enrichment of the fuel) and the geometry (heterogeneous lattice or homogeneous mixture) of the system. A few, closely packed, fairly fresh PWR fuel assemblies immersed in water can form a critical system while natural uranium alone can never become critical, no matter how large its mass, because too high a proportion of the fission neutrons are lost in nonfission reactions.

IV. DISCUSSION

A. Massive Damage of Fuel Assemblies in Pool or Reactor

Theory

During an accident where a dropped load impacts fuel, the configuration of the assemblies could be disrupted. If the disruption is extreme enough, the fuel could be crushed to the point where it fragments. This fragmentation changes the heterogeneous nature of a PWR fuel assembly and approaches a homogeneous mixture. A homogeneous mixture is by nature a less reactive system than a heterogeneous geometry, all else remaining the same (water/UO_2 volume ratio, quantity of U-235, etc.). This is mainly due to the resonance escape probability which, for a given ratio of fuel to moderator, can be increased by using a heterogeneous lattice system (such as fuel rods in an assembly). For example, in a system consisting of natural uranium (0.7 percent uranium-235) and graphite (acting as a moderator) the value of K_∞ is increased from a maximum of 0.85 in a homogeneous mixture to about 1.08 in an optimized heterogeneous (or lattice) system. The lattice spacing of a PWR fuel assembly is designed close to optimal to maximize neutron economy. Homogenization of a fuel assembly would not necessarily produce the same magnitude of change but the trend would be the same. However, with light water as moderator, it is apparently impossible to achieve a critical system with natural uranium as fuel under any circumstances. Spent fuel generally has an enrichment less than 0.9 weight percent U-235, very close to natural uranium. Therefore, there appears to be no potential for criticality of totally spent fuel that has been crushed to the point of breaking up and fragmenting, regardless of the magnitude of fragmentation and the final configuration. This cannot be catagorically stated for fuel with U-235 content above approximately 150% of natural, such as fresh or partially spent fuel. The drop in the multiplication factor due to the loss of lattice geometry (homogenization) may not be enough to overcome the higher fuel enrichment

Damage to Fuel Stored in Spent Fuel Pool

PWR assemblies are stored in racks that depend on fuel separation to ensure subcriticality. This separation produces a large water/UO_2 volume ratio that keeps the multiplication factor well below its design basis value of 0.95. If the fragmented fuel spreads out into a soup like mixture, the resulting system is even more subcritical than before. Spent fuel storage pool chemistry is typically 2000 ppm (minimum) boron as boric acid. Boron concentrations have an effect of approximately 10% $\Delta k/k$ per 1000 ppm. In Duke Power testimony before the Atomic Safety and Licensing Board for Oconee spent fuel transportation and storage at McGuire, it was stated that, assuming maximum enrichment of any assembly then in the Oconee pool (1.2%) and 2000 ppm boron, massive damage to 226 assemblies produced K_{eff} = 0.45. If the damaged fuel breaks up and falls in on itself, the

water/UO2 volume ratio may decrease to a more optimal value. If this change is large enough to overcome the loss of lattice geometry, the multiplication factor would increase. In this unlikely scenario, the 2000 ppm of boron would be more than adequate to ensure subcriticality assuming all but the freshest of fuel. If the fuel storage racks contained a solid poison (usually a boron composite), subcriticality would be maintained by its distribution through the damaged system along with the dissolved boron. The crushed stainless steel racks also act as a poison by absorbing neutrons. The stainless steel, acting in conjunction with the 2000 ppm boron dissolved in the water, will prevent criticality in the spent fuel pool of fresh or spent fuel regardless of the magnitude of fragmentation and the final configuration. If fuel and racks were crushed so compactly that all water, borated or not, was forced out, criticality would not be a concern. Light water reactor fuel needs water as a moderator of neutron energy to sustain a fission chain; lack of a moderator prevents criticality.

Damage to Fuel While in Reactor Vessel

In a reactor core, all movable control rods have a total worth of approximately 10% $\Delta k/k$ and all fixed burnable poisons have a total worth of approximately 4.5% $\Delta k/k$. Their presence has a significant effect. For light water reactor fuel to achieve criticality after massive damage, the fuel must not have seen significant burnup (fairly high enrichment) and must be moderated by the proper amount of water, i.e. the Water/UO2 volume ratio must be near optimum. Even if this highly unlikely scenario occurred, the effect would be only to boil water, not cause a steam explosion. The concern of potential criticality of massively damaged fuel can be eliminated by ensuring that a large amount of low burnup fuel cannot be affected by a dropped load and that the boron concentration is maintained at its required level.

B. Reconfiguration of Fuel Assembly and Pin Lattice

Theory

The more limiting case of a dropped load accident occurs when the fuel pins are not crushed but are pushed closer together so that the spacing of the fuel lattice is changed. In a spent fuel pool, this highly unlikely scenario would require extensive damage to the fuel storage racks while maintaining the integrity of the fuel. The assemblies must be pushed closer and the pins in each assembly close in on each other. This configuration of fuel in the storage pool resembles a reactor core, i.e. an infinite array of fuel rods. Although the pins in an assembly are already at a near optimum lattice spacing for pure water, squeezing out borated water is actually removing an absorber of neutrons so that packing pins tighter in some instances makes criticality more likely.

Damage to Fuel Stored in Spent Fuel Pool

New fuel storage pools contain no water to act as a moderator so criticality is not a problem. Optimum moderation from hydrogenous fire fighting substances are no longer considered credible scenarios in new fuel storage pools.

In testimony by Charles R. Marotta of the NRC before the Atomic Safety and Licensing Board on Oconee spent fuel transportation and storage at McGuire, he states that: a) subcriticality will be assured for spent Oconee fuel with 2000 ppm boron in the pool water and b) subcriticality will be assured for fresh McGuire fuel with at least 2000 ppm boron, if the average fuel enrichment is 2.6% U-235 or less. Duke Power testimony on the same occasion states that, assuming an enrichment of 1.2% U-235 for Oconee spent fuel and 2000 ppm boron, the pushing together of 226 assemblies produces a $K_{eff} = 0.95$. Both the NRC and Duke analyses assumed an infinite number of fuel assemblies. This is a standard calculational technique that introduces conservatism into the analysis since it eliminates neutron leakage in the horizontal direction.

The smaller the number of assemblies actually involved, the less chance of realistically attaining criticality because of increased leakage in a smaller system. Also, the less fuel (U-235) involved, the smaller the chance of attaining a critical mass. NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants", hypothesizes this scenario in Section 2.2, Criticality Considerations. This worst case configuration where assemblies and pins are pushed together into optimal reactivity volume ratios without damage to the fuel seems extremely unlikely. The fuel storage racks would prevent this configuration from occurring in a spent fuel pool.

Damage to Fuel While in Reactor Vessel

In a reactor, the boron concentration in the coolant is decreased as the fuel is burned up during the life of a cycle. This has no effect on our criticality consideration because this accident can only occur during shutdown with the vessel head off. The vessel head is not removed until the boron concentration is brought up to a prescribed level. During refueling at McGuire and Catawba, boron concentration will be $\geq$ 2000 ppm; whereas, at Oconee the concentration will never be less than 1835 ppm. Near the end of the refueling operation, the average fuel enrichment is approximately 2.6 weight percent U-235 while at the start the average enrichment is closer to 2.0. As can be seen from the attached graphs, criticality cannot be absolutely ruled out with the prescribed scenario. This goes for both a core in the reactor vessel and a core that has been discharged into the spent fuel storage pool early in its cycle. Once again, fuel enrichment, boron concentration and water/UO_2 volume play important parts.

If this scenario is deemed possible in the reactor vessel, the possibility of the core becoming critical cannot be ruled out. As stated in NUREG-0612, observing the attached graphs indicates subcriticality can be maintained by raising the boron concentration above 2500 ppm. If the worst case load drop is judged feasible, raising the boron concentration could be the solution. This, however, may require systems changes to accommodate the higher concentration.

C. Applicability of Criticality Curves to Duke Nuclear Plants

The attached graphs plot the neutron multiplication factor as a function of the Water/UO2 volume ratio for Westinghouse 15 x 15 fuel. Oconee uses Babcock and Wilcox 15 x 15 fuel while McGuire and Catawba use Westinghouse 17 x 17 fuel (Catawba is scheduled to use an "optimized" design variation of the Westinghouse 17 x 17). Numerous comparisons were made of the three different fuels to determine their relative reactivities. The computer code OCELOT was used to model infinite arrays of the various fuel rods in order to calculate the respective neutron multiplication factors. The analysis showed the three fuels to behave virtually the same from a nuclear criticality standpoint; no significant difference exists. The very slight variations in calculated neutron multiplication factors show fuel used by Duke to be slightly more convervative than Westinghouse 15 x 15 fuel and thereby bounded by the plotted curves. However, any and all differences are slight enough that the three types of fuel can be considered the same.

V. CONCLUSIONS

Recriticality of damaged fuel is dependent on a number of factors but the following statements can be made:

1. In general, recriticality in a spent fuel pool is not a problem if dissolved boron is present in the required amount and the storage racks are included.

2. Subcriticality is ensured if the damaged fuel is totally spent (close to natural enrichment).

3. Damage to fuel in a reactor core must be studied closely to determine the potential for criticality.

4. Increasing the boron concentration during shutdown from 2000 ppm to above 2500 ppm would ensure subcriticality in the reactor vessel.

VI. REFERENCES

1. "Nuclear Reactor Engineering" by Samuel Glasstone and Alexander Sesonske.

2. Duke Power Company, Amendment to Materials License SNM-1773 for Oconee Nuclear Station Spent Fuel Transportation and Storage at McGuire Nuclear Station, Docket No. 70-2623, Affidavit of S. B. Hager

VI. REFERENCES (Continued)

3. Duke Power Company, Amendment to Materials License SNM-1773 for Oconee Nuclear Station Spent Fuel Transportation and Storage at McGuire Nuclear Station, Docket No. 70-2623, Testimony of Charles R. Marotta.

4. "Recriticality Potential of TMI-2 Core", by C. R. Marotta.

5. NUREG/CR-2155, "A Review of the Applicability of Core Retention Concepts to Light Water Reactor Containments", Sandia National Labs, September 1981.

6. NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants".

7. Memo to File, "Criticality Analysis", October 2, 1979, by Norman T. Simms. File: MC-1201.28, CN-1201.28, P81-1201.28.

8. McGuire Nuclear Station, Technical Specifications.

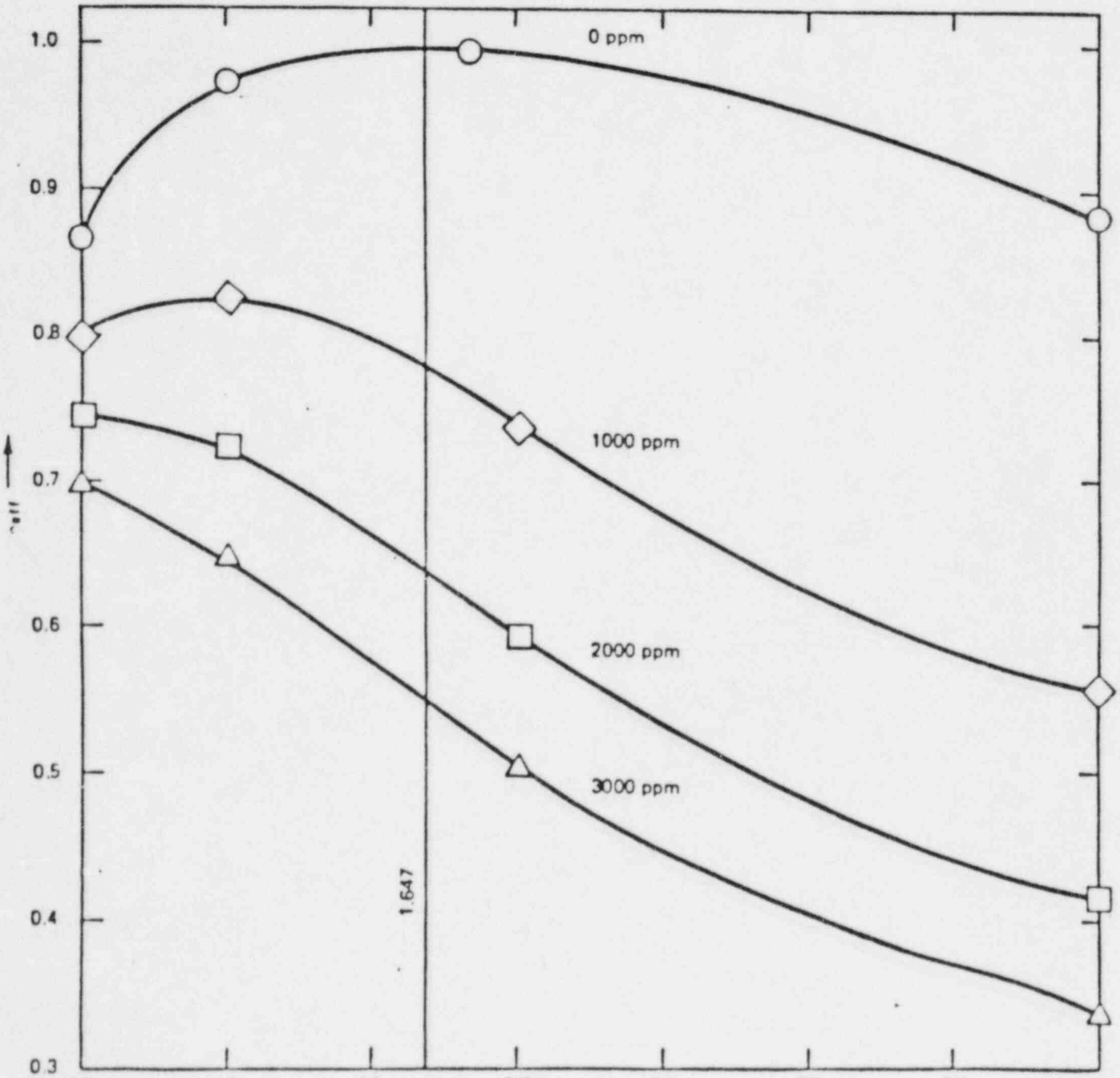
9. Catawba Nuclear Station, Technical Specifications.

10. Oconee Nuclear Station, Technical Specifications.

Attachments

Parameters are as follows. (The dimensions used are those of Westinghouse 15 x 15 fuel)

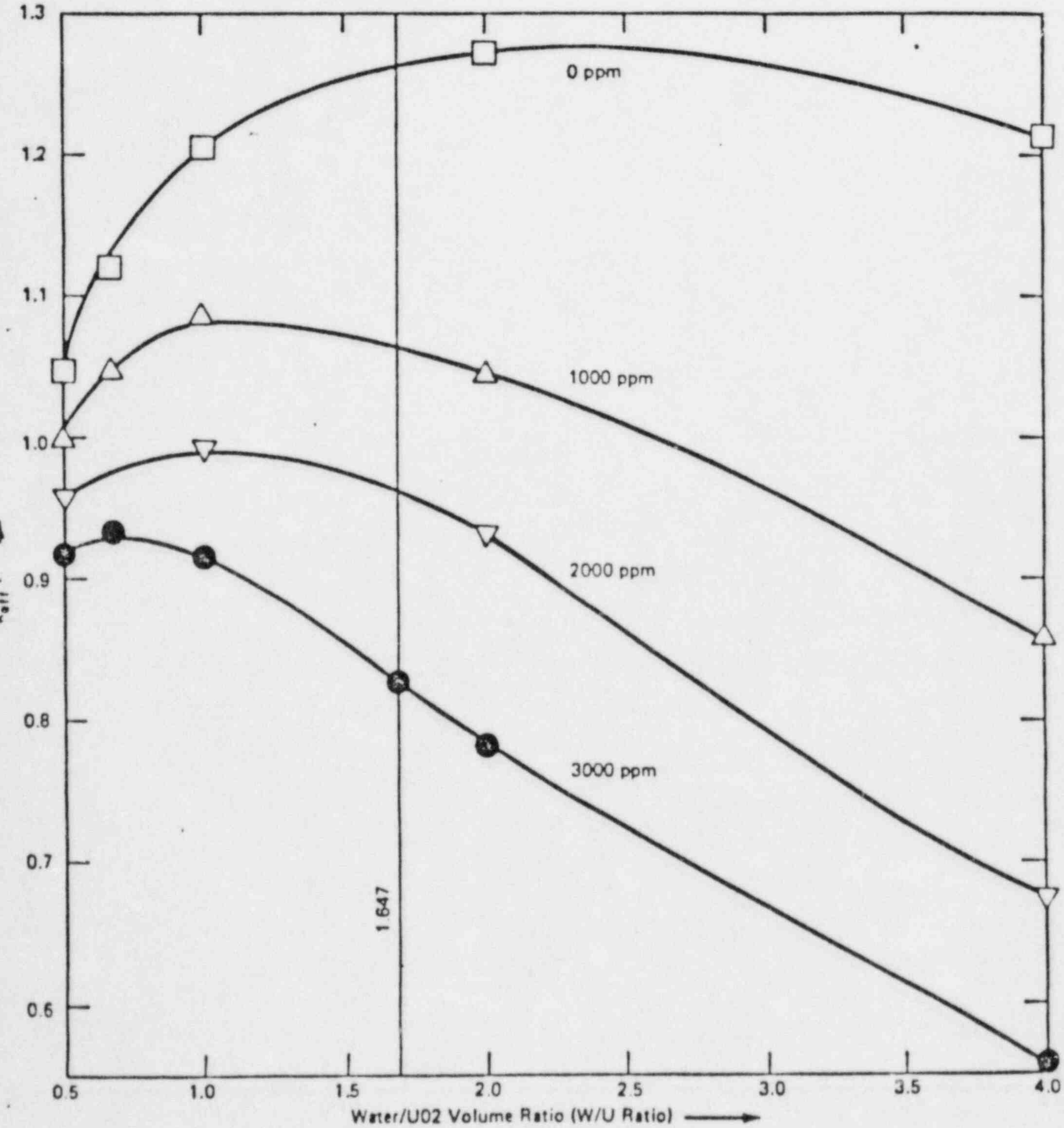
Fuel Pellet Diameter	0.3659"
Zirc Clad Inside Diameter	0.3734"
Zirc Clad Outside Diameter	0.4220"
As-Built W/U Ratio	1.647
Temperature	20 DEGC
Fuel Material	0.9 w/o U235



NEUTRON MULTIPLICATION FACTOR FOR INFINITE ARRAY OF FUEL RODS IN BORATED WATER

Parameters are as follows. (The dimensions used are those of Westinghouse 15 x 15 fuel)

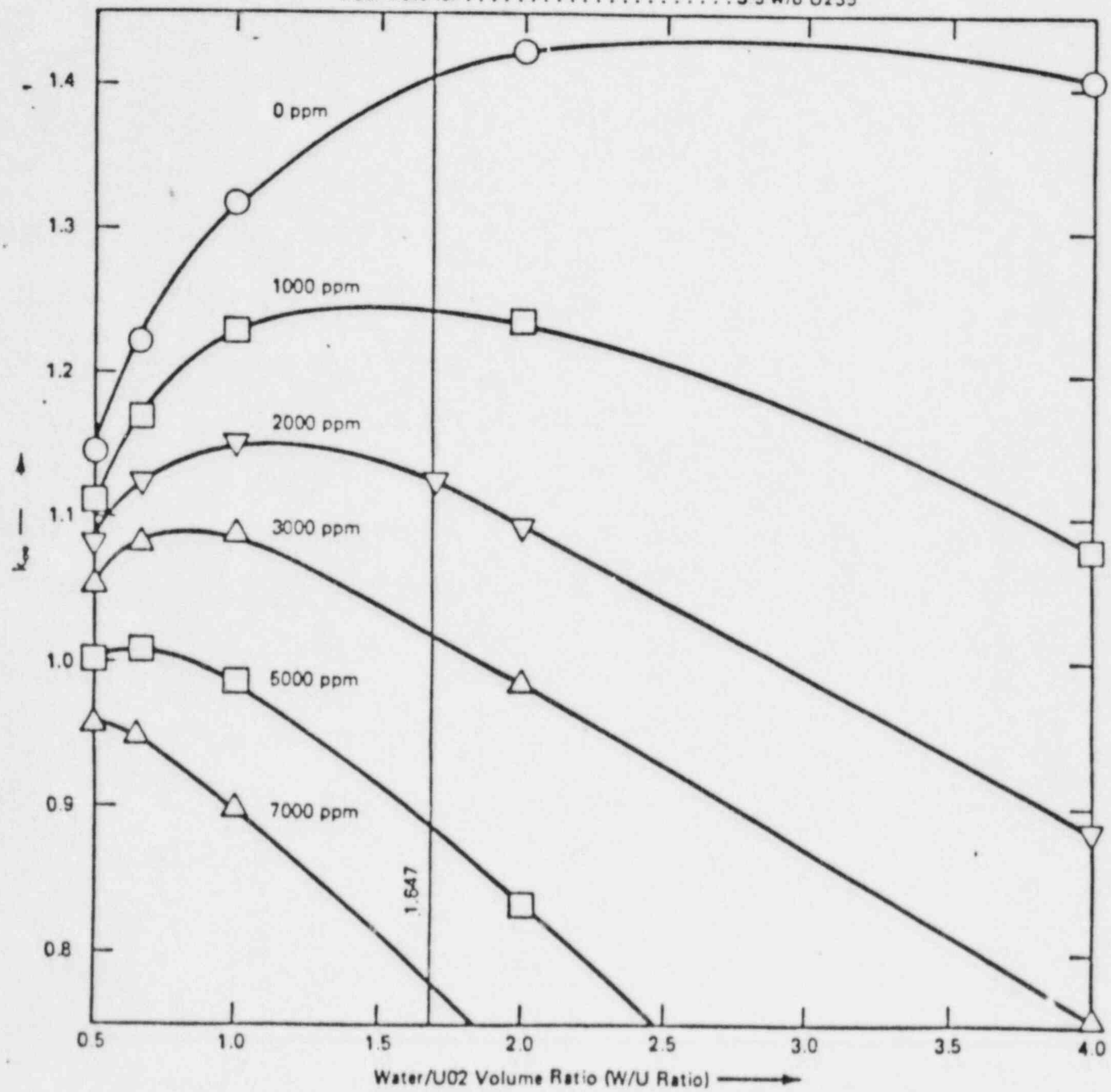
Fuel Pellet Diameter	0.3659"
Zirc Clad Inside Diameter	0.3734"
Zirc Clad Outside Diameter	0.4220"
As-Built W/U Ratio	1.647
Temperature	20 DEGC
Fuel Material	2.0 w/o U235



NEUTRON MULTIPLICATION FACTOR FOR INFINITE ARRAY OF FUEL RODS IN BORATED WATER

Parameters are as follows. (The dimensions used are those of Westinghouse 15 x 15 fuel)

Fuel Pellet Diameter	0.3659"
Zirc Clad Inside Diameter	0.3734"
Zirc Clad Outside Diameter	0.4220"
As-Built W/U Ratio	1.647
Temperature	20 DEGC
Fuel Material	3.5 w/o U235



NEUTRON MULTIPLICATION FACTOR FOR INFINITE ARRAY OF FUEL RODS IN BORATED WATER

SUPPLEMENTAL EVALUATION OF IN-VESSEL CRITICALITY

FOR

NUREG-0612, "CONTROL OF HEAVY LOADS AT NUCLEAR POWER PLANTS"

McGUIRE NUCLEAR STATION

According to Section 4.2.2 of Appendix A, the licensee can demonstrate that crushing the core will not drive it critical by using the core refueling neutronics analysis for uncrushed fuel and showing that k_{eff} for an uncrushed core is no greater than 0.90. Then, using the estimated 0.05 maximum reactivity insertion due to crushing, the maximum achievable k_{eff} is still less than 0.95. We will show compliance with the 0.95 limit using McGuire specific fuel parameters.

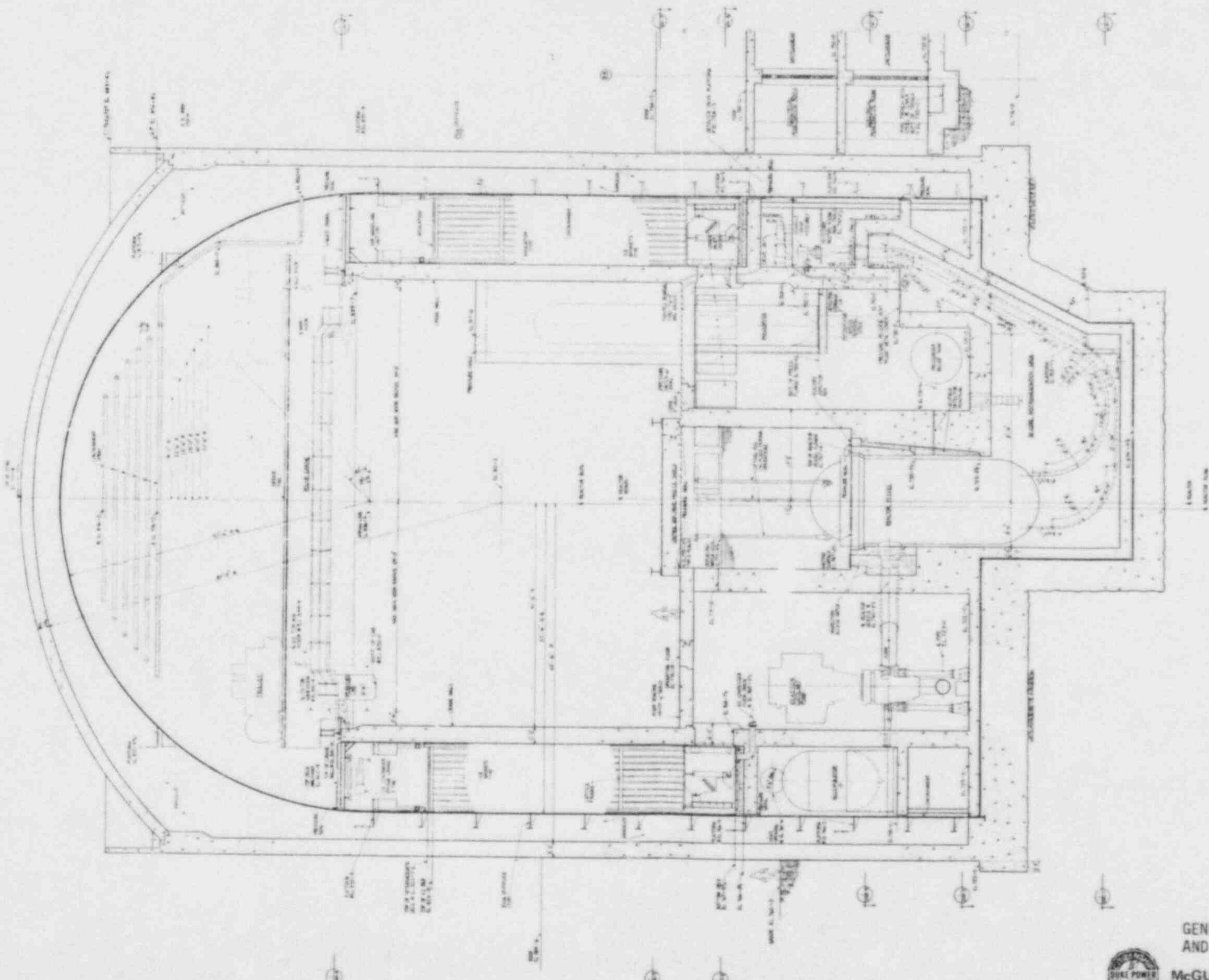
From Table A.1 of WCAP-9323, "The Nuclear Design and Core Physics Characteristics of the W. B. McGuire Unit 1 Nuclear Power Plant Cycle 1," we see that during fuel loading (ambient temperature and pressure) with all control rods in, a boron concentration of 1367 ppm gives K_{eff}=0.95.

McGuire Technical Specifications states that the boron concentration will be maintained at a minimum value of 2000 ppm during refueling. To account for the difference in reactivity between 1367 ppm boron and 2000 ppm boron, we refer to Figure A.10 of WCAP-9323. For BOL, 68°F an average value for the differential boron worth is -11.95 pcm/ppm (this assumes linear interpolation of the curve out past the 1500 ppm end point). Thus;

$$(2000ppm-1367ppm)(-11.95pcm/ppm)(10^{-5}\Delta k/k/pcm)=-.0756\Delta k/k$$

$$K_{eff} = 0.95-.0756 = .8744$$

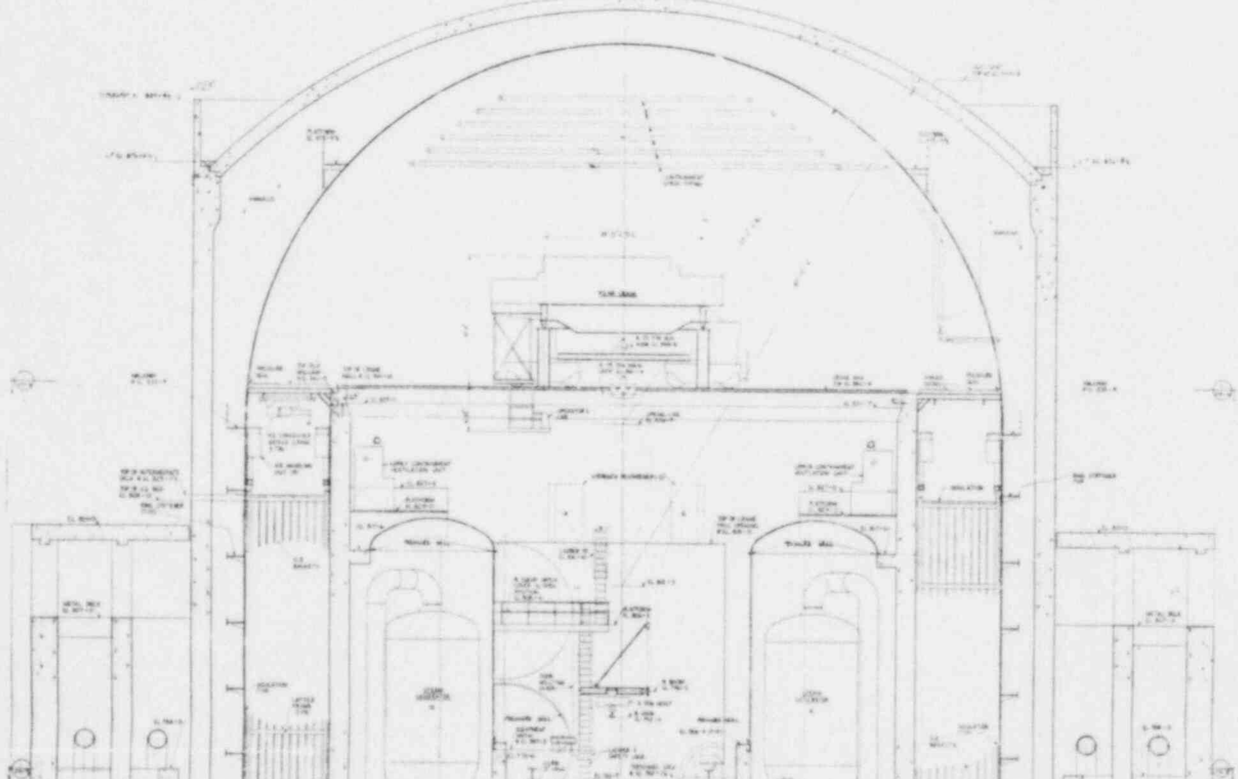
The calculated value of k_{eff} is well below the 0.90 limit given.



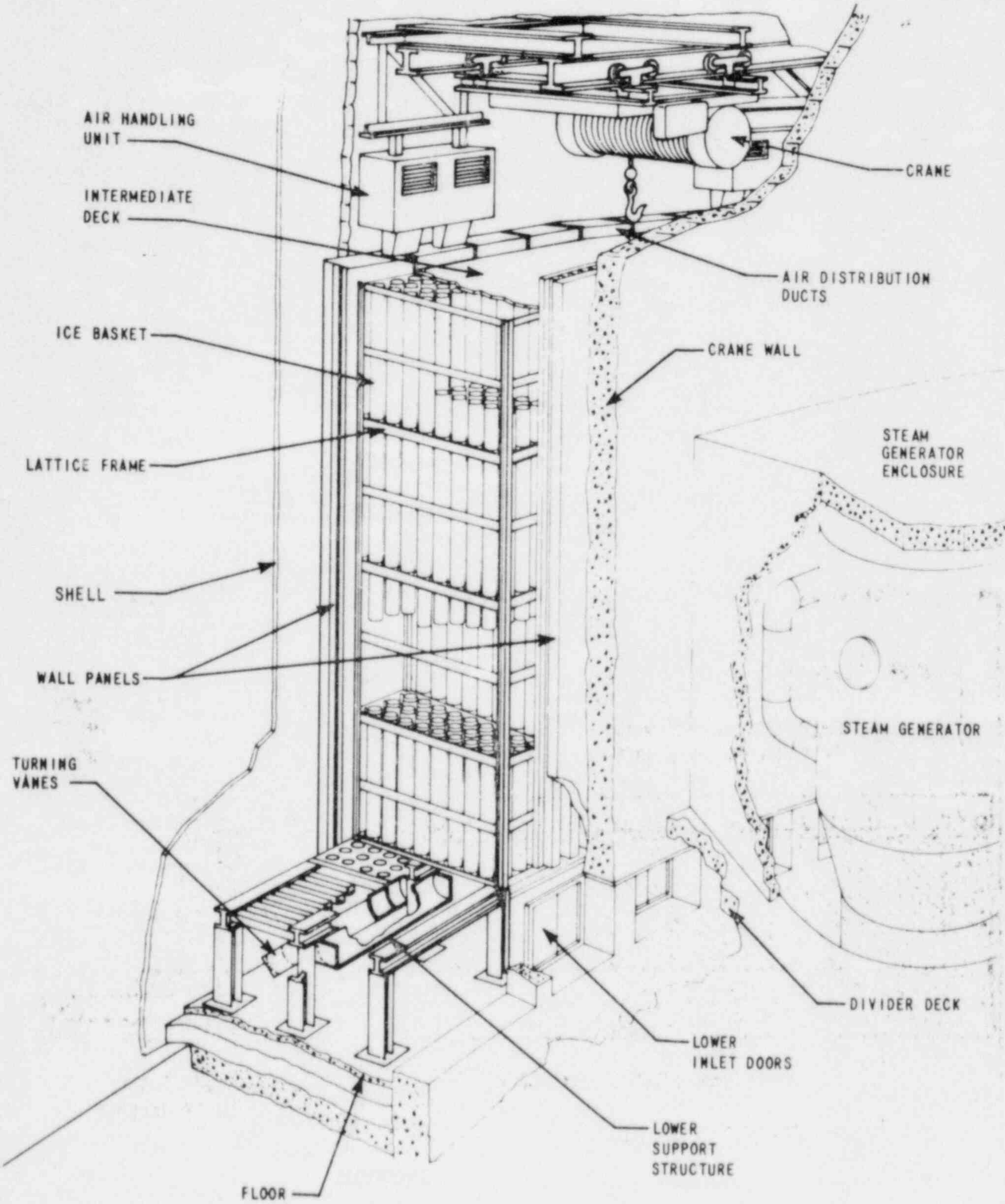
GENERAL ARRANGEMENT CONTAINMENT
AND REACTOR BUILDING SECTION

McGUIRE NUCLEAR STATION

Figure 1.2.2-14
Revision 5



GENERAL ARRANGEMENT CONTAINMENT
AND REACTOR BUILDING SECTION
McGUIRE NUCLEAR STATION
Figure 1.2.2-15
Revision 5



ISOMETRIC OF ICE CONDENSER

McGUIRE NUCLEAR STATION

Figure 6.2.2-1