GLOBAL ENERGY CONSULTANTS, LLC

DOCKET NUMBER PR 72
PROPOSED RULE (68FR 49683)

September 11, 2003

(1)

DOCKETED
USNRC

September 12, 2003 (9:54PM)

OFFICE OF SECRETARY
RULEMAKINGS AND
ADJUDICATIONS STAFF

Secretary
U.S. Nuclear Regulatory Commission
Washington DC 20555-0001

ATTN: Rulemaking and Adjudications Staff

Reference: Federal Register RIN 3150-AH26
Volume 66, No. 160
August 19, 2003

Attachments: Appendix A: Professional Information on Mr. Thomas A. Jones
Appendix B: Marked Pages from NUHOMS FSAR

Subject: Direct-to-Final Rule on Amendment No. 5 to CoC No. 1004

Dear Sir:

The above referenced proposed direct-to-final rule seeks to amend the regulations to revise the Standardized NUHOMS cask system to incorporate NUHOMS-32PT DSC under Amendment No. 5 to CoC Number 1004. We have examined the underlying technical material (the revised Technical Specification, the SER, and TN West's FSAR) and have determined that:

i. It appears that the NRC has not considered several essential technical issues that bear upon public health and safety.

ii. The FSAR submitted by the CoC holder is inadequate in its treatment of certain key issues that affect public health and safety.

iii. The NUHOMS 32-PT DSC fails to pass certain basic tests as a fabricable and usable dry storage canister.

Our comments provided hereunder with appropriate citations from the publicly disclosed material should convince the Commission that the proposed rule will be noncompliant with NRC's published regulations, NUREGs and ISGs and, therefore, adverse to public interest.

As my professional profile provided as Appendix A to this letter would indicate, I have over thirty years of experience in the commercial nuclear industry. I have not participated in any intervention groups in the past, nor am I a part of any ongoing anti-nuclear action. My professional career has been entirely dedicated to enhancing the reliability of commercial nuclear power. I have had no commercial relationship with any CoC holder at any time in my professional career. The sole thrust of my comments is to request the NRC to probe the technical aspects of the proposed change in greater depth than the commission seems to have done thus far, for both the reputation of the agency and the public's confidence in the robustness of dry storage systems may be shaken by accepting a design that, based on the publicly available information, appears to be deficient and inadequate.

Template = SECY-067

SECY-02

BASES FOR ASSERTION OF DEFICIENCY

1. Ambiguous Technical Specification with Respect to Acceptable Seismic Motion

Page A-2, ref. [1]* sets the limit of 0.17g vertical and 0.25g horizontal on seismic accelerations.

Unfortunately, these limits are identified as *site-specific parameters.* The SER (reference [2])* is equally ambiguous, stating in para. 3.1.2.1.7

> "The design earthquake for the NUHOMS 32PT DSC system is based on an earthquake that produces a horizontal ground accn of 0.25g and vertical accn of 0.17g..."

An inspection of the CoC holder's FSAR (ref. [3]*, para. M.3.7.3) reveals that 0.25g and 0.17g are applied as peak horizontal and vertical ground accelerations on the NUHOMS system. It is a common knowledge in geo-mechanics that the free field accelerations at the site can be magnified considerably on the pad due to soil-structure interaction (SSI) effects.

TNW's analysis of NUHOMS assumes that 0.25g and 0.17g horizontal and vertical accelerations are applied on the HSM basemat; these are thus the limiting values of *on-the-pad accelerations,* not "site parameters" as noted in the Technical Specification.

Recommendation:

> The Technical Specification should be corrected to state unequivocally that 0.25g and 0.17g are, respectively, the maximum permitted values of the peak horizontal and vertical accelerations at the NUHOMS/ISFSI pad interface.

2. Lack of Demonstration of Compliance with §72.130

§72.130 mandates that the ISFSI must be designed for decommissioning, particularly, it must be designed "to facilitate the removal of radioactive wastes..."

Based on the information presented in the FSAR [3,4], NRC's SER, one cannot conclude with reasonable confidence that the loaded 32PT DSCs will be able to be removed by the hydraulic ram, after the NUHOMS modules have been on the storage pad for their licensed life (twenty years). There are two main technical reasons that would make an experienced engineer pessimistic with regard to the removability of the loaded DSCs after twenty years of storage. They are:

* References are listed at the end of this letter.

i. Potential for long-term settlement of the pad, and

ii. Weathering (corrosion) of the DSC/rail interface under extended exposure (twenty years) to the elements.

Long-Term Settlement

Paragraph 1.2.9 [1] stipulates that the transfer "cask must be aligned with respect to the HSM so that the longitudinal centerline of the DSC in the transfer cask is within 1/8 inch of its true position when the cask is docked with the HSM front access opening".

This requirement, imposed to enable the DSC to be moved horizontally, is tedious, but doable during initial loading. However, calculations performed for typical storage pads loaded with heavy casks show that the long-term differential settlement from soil creep can be several inches over twenty years. NUHOMS's FSAR [4, para. 1.2.5] makes no special demands on the soil strength to limit long-term settlement of the pad.

> "Site Soil Conditions: The site soil conditions required for a NUHOMS® ISFSI are comparable to those needed for metal or concrete vertical cask storage systems. As the basemat and foundation are not important to safety, commercial industry practices are utilized to ensure that adequate conditions of the site sub-grade are provided."

There are no specific strength limits applied on the NUHOMS pad either (as is the case for other certified systems such as NAC-UMS) which, along with the absence of a mandated hard subgrade, would likely lead to several inches of differential settlement of the pad over twenty years of storage, and the user's ability to maintain the alignment specified in [1, para. 1.2.9] will be lost. The DSC will be in an irremovable state, in direct violation of 10CFR72.130.

Effect of Weather

Under the general CoC authority, the NUHOMS system can be installed at any site in the U.S., including coastal sites and marine environments. The potential for surface corrosion, including pitting of the DSC and HSM rail surfaces under the ambient environmental conditions and its effect on the removability of the DSC, has not been considered in the FSAR [4] or NRC's SER [2]. This is in violation of 10CFR72.236(m).

Surprisingly, the maximum allowable hydraulic push-and-pull forces specified in the FSAR are not equal. The push force is 80 kips; the permitted pull force is only 60 kips. As discussed above, it is during the removal of the DSC when the DSC must be dragged over the corroded HSM rails, that the risk of failure to remove the canister lies. Yet, the allowable pull for the DSC extraction

condition is 25% less than the available push force during initial insertion! Further, the coefficient of friction during DSC push assumed in the FSAR to be 0.2 is unrealistically low for weathered sliding surfaces.

Also, in the FSAR, we do not see any stress analysis of the DSC bottom cover plate that is being pulled by the hydraulic ram against friction with internal pressure present in the canister. Internal pressure and the hydraulic ram pull force will act in concert to maximize the stress level in the cover plate and its junction with the DSC shell. Neglect of analysis of this scenario leaves the structural adequacy of the bottom outer lid open to question.

3. Lack of Compliance with 10CFR72.122(b)(2)(1)

10CFR72.122(b)(2)(1) requires the structures, systems and components to be able to withstand the effects of natural phenomena such as earthquakes.

The process of inserting a DSC in the HSM requires careful alignment of large fabricated components in open air. The time duration for such activities can be long. The NRC imposes seismic requirements on canister transfer outside of Part 50 structures even in vertical operations (see NAC-UMS or HI-STORM FSAR, for example). Yet, for the more tedious horizontal insertion process in NUHOMS, there is no treatment of a concurrent seismic event or even tornado-borne missiles during DSC transfer operations. This violates §72.122(b)(2)(1).

Further, it is noted that the 32PT DSC is the heaviest canister proposed for use thus far in the HSM. NUHOMS's FSAR asserts that the DSC support structure is braced, presumably to incorporate seismic resistance. A review of the sketches provided in the FSAR show no bracing.

In Appendix B, I have placed a marked copy of the pages from the FSAR [4], where I have marked (in felt pen) the missing braces.

Without the braces, the DSC support structure in the HSM is weak against axial or lateral overturning moments, especially the increased g-loads that will accompany the heavier 32PT DSC.

The consideration of the tornado-borne missile in the FSAR [4] is equally oblivious to the real vulnerability of the HSM. The entire three-foot thick top roof is held by a mere four anchors about 1-1/2 inches diameter and the concrete-filled front door (over 7,000 lbs. in weight) is not even held by bolts (rather, by three straps). The FSAR [4] provides no analysis of the integrity of these weak locations in the HSM under natural environmental phenomena loads. (How these structural features will resist a larger impact such as a plane should also be a matter of concern to the agency in the after-9/11 world.)

4. Lack of Compliance for Postulated Accident Events per §72.122(b)

According to the FSAR [3,4], the DSC 32PT has purportedly been analyzed for a drop from 80 inches onto an essentially unyielding surface with the added assumption that the TC is rigid. This event is postulated to account for a potential drop of the loaded DSC in the TC during its handling on the basemat [2, page 3-12]. The calculations to compute the g-load, however, use an antiquated method [3, M.3.7.10 and 4, Section 8.2.5], which was determined to be unconservative by the NRC in the mid-1990s.

The NRC paper [5] sets down the acceptable method for reliably and conservatively predicting the g-load. The method relied on in the FSAR, in the opinion of this writer, is unconservative: a much higher value than 75g's will develop if the DSC 32PT undergoes a free fall of 80 inches on a rigid surface without the benefit of an impact limiter.

5. TC/DSC Lift Height Per Technical Specification Para. 1.2.13

The Technical Specification permits lift height of 80 inches in cold (temperature between 0 - 20°F) conditions based on NTD temperature considerations of TC's materials. The underlying documents (SAR or SER), however, do not address the top and bottom shield plugs that are very thick (over 6 inches) and made of a steel that is low-temperature incompetent (A-36). At -20°F, I reckon that the A-36 steel plugs will suffer extensive fracture under 75g's impact, perhaps even pulverization.

Of an even greater concern is the clear absence of critical structural welds in the fuel basket in DSC 32 PT. I have manually circled areas in the limited drawing details released to the public that show absence of welds in the fuel basket at critical load transfer locations under a horizontal drop condition (see Appendix B). TNW's stress analysis of the basket appears to have a serious error – perhaps an erroneous assumption in the finite element model. Critical stress analyses figures are deleted from the non-proprietary copy; therefore, I cannot provide further help.

6. Absence of Non-Mechanistic Tipover

NUREG-1536, Chapter 11, V.1 states that "an event may be analyzed for regulatory purposes even though no credible cause can be identified. Such events should be clearly identified as non-mechanistic."

NRC's regulatory practice has been to require a "non-mechanistic tipover" analysis of casks in long-term storage. Accordingly, to the NUHOMs FSAR [4], each HSM is *free-standing*. The height (15 feet) to width ratio (9.7 feet wide) of the HSM is comparable to vertical ventilated systems (that tend to be about 18 feet high by 11 feet diameter) where NRC has always

demanded a non-mechanistic tipover analysis. Why the special dispensation for NUHOMS, with its top-heavy structure (a 3-foot thick top roof held in place by slim anchors)?

7. Questionable Compliance with 10CFR72.124 (Criticality Safety)

The neutron absorber panels in 32PT DSC appear not to be "fixed" as required by §72.124(b). Further, the required B-10 loading in the neutron absorber panels is miniscule (merely 0.007 gm/sq. cm., less than even 52BT for BWR fuel (which is 0.016 gm/sq.cm.), and a small fraction of that used in other casks (such as NAC-STC). The reliance for reactivity control seems to be based on the so-called Poison Rod Assemblies (PRAs). These PRAs, vital to criticality control, are little more than stainless steel tubes filled with "B_4C pellets" (see [2], para. 3.1.4.2). There are no requirements imposed on the size and integrity of the welds that will join the closure plugs to these thin-walled tubes (as little as 0.018 inch thick per Figure M.1.6-2 in [3].

The so-called non-structural PRA closure welds, without any regulatory requirements on their NDE, are the sole barrier against leaching out Boron Carbide from the PRAs! Considering that the PRAs will be subject to thermal stresses during fuel loading and shall be quite hot in long-term storage, a total reliance on the micro-seal welds to hold B_4C in place to preserve criticality safety appears to be incredulous to this writer. Also, there is no requirement to purge air and moisture from the PRA tubes before seal welding its contents. This means entrained air and moisture will be locked in every PRA in the stored fuel!

8. Non-Compliance with §72.236(h)

§72.236(h) states that the "spent fuel storage cask must be compatible with wet and dry spent fuel loading and unloading facilities": The DSC 32PT canister is in violation of this requirement.

The storage slots in the DSC 32PT are 8.7" x 8.7" (*nominal*) opening [2, page 3-2]. The FSAR [4, page 1.3-2] specifies "the minimum open dimension or each fuel compartment is 8.60" x 8.60". Having worked for PWR NSSS suppliers for a long time, I can state with certainty that no Westinghouse or B&W plant has fuel storage racks with 8.6" (min.) or 8.7" (nom.) opening dimension. Irradiated fuel tends to bend, bow and twist in the reactor; for this reason, PWR reactor suppliers require large storage cell openings. The 32PT DSC, with 8.6" (min.) opening would be a stuck fuel event engineered to happen.

In a related matter, I must express deep reservation about the loose aluminum blocks (visible in Figure M.3.7.3 in [3]) that are assumed to be snugly fitting. DSC 32PT will be made from a thinner shell (1/2") (to hold a heavier basket) than prior NUHOMS's DSCs (5/8" thick shell). This means that the shell in DSC 32PT will ovalize more from its dead weight and from full-length butt welds. Snugly fitted aluminum blocks may appear acceptable on paper, but in real hardware they will be impossible to manufacture. Recall that lack of fabricability of VSC-24

baskets (cracking of steel plates at the toe of the bend) caused the industry an untold amount of grief.

9. Non-Compliance with §72.122(l)

I am utterly surprised to learn from the supplier's FSAR that a loaded DSC 32PT canister will have no provision to be lifted on its own: It must be lifted by the TC. If the DSC were to be separated from the TC under an accident event, there will be no means to lift and handle the canister!

I consider the lack of ability to separately handle a loaded canister to be a severe weakness that violates the notion of retrievability under §72.122(l).

10. Non-Compliance with an Invoked ISG (ISG-11, Revision 2) and §72.122 (h)(1)

Page A-50, Section 1.2.24 of the Technical Specification [1], states, "... for the NUHOMS-32PT system, the fuel cladding limits are based on ISG-11, Revision 2." Not so, according to the publicly released information. I quote from page 2 of ISG-11, Rev. 2:

> "Accordingly, the materials reviewer should coordinate with the thermal reviewer to assure that the maximum calculated temperatures for normal conditions of storage, and for short-term operations including cask drying and backfilling, do not exceed 400°C (752°F)."

In direct violation of the above requirement, the FSAR [3, Section 4.1] states:

> "During short-term conditions, the fuel temperature limit is 570°C".

Calculated temperature values in Table M.4.2 indicate that the ISG-11 Revision 2 limit is indeed exceeded by wide margins under short-term normal conditions.

11. Material Selection: Conformance to 10CFR72.122(a)(b) and (c)

Use of durable materials that are proven for their intended function must be a basic plank of dry storage system design, and a mandated fact under the above-cited Part 72 paragraph. I find two major objections to the materials being proposed for DSC-32PT.

First, the shield plugs at the two ends of the DSC are made from one of the cheapest carbon steel available (A-36). Note that the lower plug (along with air) is permanently sandwiched between the two stainless plates. This plug will expand and contract under heat, as will the entrained air in the space, constantly stressing the welds that confine it. Thermal differential expansion between

carbon steel and stainless steel will further increase stresses in those same welds. Why could not the plugs be made of machined stainless steel, which would eliminate material incompatibility, remove most entrained air, and thus and remove long-term concerns?

Second, is the neutron absorber. I have been unable to locate any specificity on the brands of neutron absorbers permitted by the CoC. Neutron absorbers use aluminum, which is a most reactive material. NRC has been wise in controlling the specific make of neutron absorbers that are permitted to be used. This caution is well placed, considering the 1996 hydrogen ignition event in SNC's product. I am quite surprised to find [2, page 3-9] that even purging of the canister during lid welding is not required. I believe it is unsafe to make purging elective if aluminum-based neutron absorber coated carbon steels are present in the canister. Recall th e lesson learned from the Columbia Generating Station experience!

I strongly recommend that the CoC *specify* the acceptable neutron absorbers to ensure compliance with the above-cited regulation and not let a CoC holder make the choice of neutron absorber unilaterally.

12. Potential for Fire in the Vicinity of the HSM Ignored (Incorrectly)

Referring to Paragraph M.4.6.3 of the FSAR [3], we conclude that a fire event in the vicinity of the HSM has been ruled out. This inference is also supported by the text matter in reference [4]. The FSAR statements ruling out fire around the HSM are erroneous because the hydraulic fluid in the ram and the fuel in the heavy-haul trailer are credible sources of fire for a previously loaded HSM located in the vicinity of the HSM being loaded.

The a' priori exclusion of fire analysis at the HSM is inconsistent with NRC's previous certification reviews of other ventilated systems; it is also unsafe.

13. Violation of ASME Code Requirement

Section M.3.1.2.1 of the FSAR [3] states that the inner bottom cover plate-to-shell joint is subjected to volumetric and liquid penetrant examination as required by Subsection NB of the Section III of the ASME Code.

This weld cannot be radiographed or U.T. examined by virtue of its geometry.

14. Unsafe Practice for Canister Reflooding

The SER [2, Section 4.8] accepts sudden quenching of irradiated fuel at 678°F in water during reflooding operation. Quenching would cause a sudden cooling of the fuel and the 117°F

temperature limit would undoubtedly be exceeded, a restriction imposed by ISG-11, Revision 2, presumably to protect semi-brittle irradiated fuel from thermal shock.

I urge the NRC to reconsider this unnecessary regulatory leniency.

15. Failure to Analyze a Credible Flood Event

The consideration of flood in the FSAR is merely to treat it as a source of hydrostatic load [3, Section 3.7]. A low elevation flood that submerges the bottom duct is far more dangerous. A partially submerged HSM, heated by the DCS through radiation and convection and chilled by the rising floodwaters will cause severe thermal stresses in its reinforced concrete structure. Because the HSM's walls are both structural members and biological shield, a thru-thickness crack from large thermal strains induced by a short-duration flash flood will be unacceptable for public health and safety.

There is no consideration of this scenario in the supporting licensing material provided by TNW: It calls for a careful analysis.

General:

I am surprised and disappointed that the CoC uses a product designation name like "-32PT," where the "T" stands for transportable; and uses the words, " ... and T is to designate that the DSC is intended for transportation in a 10CFR71 approved package," when this CoC pertains only to storage. I can say from personal experience, that foreign utilities in particular, do not always recognize the distinction. I therefore question the purpose for using this designation, or making this statement.

REFERENCES

[1] Attachment A, Technical Specification to CoC, USA/72-1004

[2] Preliminary Safety Evaluation Report, CoC No. 1004, Amendment No. 5.

[3] 72-1004 Amendment No. 5 to FSAR, TN West, June 2001, Rev. 0.

[4] FSAR for the Standardized NUHOMS System, TN West, August 2000.

[5] "NRC Staff Technical Approach for Spent Fuel Storage Cask Drop and Tipover Accident Analysis" by D.T. Tang, G. Raddatz and F.C. Sturz, USNRC, Washington, DC (1997).

I trust that the NRC will find the above comments to be helpful in safeguarding public health and safety.

Sincerely yours,
Global Energy Consultants, LLC

Thomas A. Jones
President

THOMAS A. JONES

944 Main Road, Granville, MA 01034 USA   413-357-6609   thomasajones@comcast.net

A senior professional with extensive experience in developing and managing large nuclear power businesses on a global scale. Experience includes all aspects of domestic and foreign business development strategies, negotiations with private and government officials, and contract execution with domestic and foreign clients. Total business volume: approximately $1B.

EDUCATION

Rensselaer Polytechnic Institute, Troy, NY – BSME
Tau Beta Pi, Pi Tau Sigma, Noelle Callahan Award, RPI Medal

Massachusetts Institute of Technology, Cambridge, MA – MSME
National Science Foundation Fellowship

PROFESSIONAL EXPERIENCE (reverse order)

Global Energy Consultants, LLC — **December 2000 - Present**
President

Established a consulting firm of 15 people supporting worldwide commercial nuclear power businesses, using only proven, senior experts in their field, each with no less than 25 years experience. Capabilities cover all technical areas associated with a typical NSSS (Nuclear Steam Supply System) vendor.

Asea Brown Boveri, Inc./Westinghouse Electric Company LLC — **1999 – 2000**
Director, Business Development

Responsible for developing a new business line at Newington Operations, Newington, NH, involving the fabrication of MPC's (Multi-Purpose Canisters) used by the commercial nuclear industry for storage and transportation of spent nuclear fuel. These are large, precision stainless steel components – sometimes using containing special materials for neutron absorption, which must be designed and fabricated to strict quality requirements.

- Developed a fully comprehensive business plan extending over ten years, with an IRR of 28%, and two year return on investment.
- Specified required facility throughput, required investment, and required facilities expansion at Newington.
- Achieved business volume: approximately $20M/year
- Separately by request of the president of BNFL, Inc., critiqued the long-term business plan of BNFL Fuel Solutions, Inc., a U.S. designer of dry storage cask equipment.

Asea Brown Boveri, Ltd., Zurich, Switzerland **1994 – 1998**
Director, European Nuclear Projects and Business Development

Directed all business development and project activities aimed at developing business in the Former Soviet Union (FSU), representing the ABB, Ltd., Nuclear BA (Business Area) which was comprised of ABB-CE, ABB TRC (Sweden), ABB Reaktor (Germany), ABB Atom (Sweden), and ABB Barras Provence (France). Reported directly to the Chairman, ABB Nuclear Power. Initial, primary focus was on Ukraine, which is ranked seventh in the world in dependence on nuclear power.

- Negotiated and established a contract with the U.S. Department of Defense to create a commercial joint venture company between ABB-CE and Monolit, Kharkov, Ukraine, to supply instrumentation and control systems to former Soviet operating nuclear plants in order to substantially improve safety. Monolit is an FSU missile guidance systems manufacturer. The resultant company, ABB-Monolit, was established, technology was transferred to the company, and the entity became a success viable commercial endeavor.
- Established a Four-Phase Nuclear Development Program between the U.S. and Ukraine. This program received endorsement by: the Ukrainian Parliamentary Committee on Nuclear Policy and Safety; Goscomatom (Ukraine's version of the U.S. DOE); the Ukrainian Ministry of Energy; the Deputy Prime Minister of Ukraine; Mr. Percy Barnevik, President and CEO of ABB, Ltd.; Director Nuclear Power, U.S. DOE; and Director Nuclear Power, U.S. DOS. The program included all aspects of nuclear power from nuclear I&C upgrades (mentioned above), to nuclear plant completions (Khmelnitsky-2 and Rovno-4), and to new nuclear plant construction.
- Negotiated an agreement between ABB, Ltd., EGL (a Swiss electricity trader) and Ukraine to sell $100M/year worth of electricity to the West European UCPTE grid, in exchange for placing 30% of this amount in escrow (Swiss bank) to effect nuclear plant upgrades solely by ABB, Ltd.
- Guided ABB, Ltd. to sole source status with the Ukrainian government to complete Khmelnitsky-2 and Rovno-4. Estimated value: $1.4B.

Asea Brown Boveri, Inc. **1992-1994**
Project Director, Advanced Reactor Design, NPR-MHTGR

Directed all work associated with the development of major reactor component design and fabrication for the U.S. DOE's 1000 MWe MHTGR (Modular High Temperature Gas-Cooled Reactor). The mission of this reactor design was to produce tritium, an essential ingredient for nuclear weapons, and electricity.

- Directed all ABB-CE activities, in concert with Stone and Webster, Burns and Roe, and General Atomics for the CEGA, Inc. under contract with the U.S. DOE. This project was DOE's top priority for three years.
- Managed a staff of 135 full time ABB engineers at three locations: Windsor, CT; Chattanooga, TN; and San Diego, CA.
- Managed cost and schedule in accordance with the U.S. government's CSCS (Cost and Schedule Control System). Completed ABB-CE's scope, which included engineering and full scale testing, within $1000 of the original budget.
- Contract value: $400M.

Combustion Engineering, Inc. & Asea Brown Boveri, Inc. **1990-1991**
Project Director, Advanced Reactor Design, SIR (Safe Integral Reactor)

Led a consortium comprised of ABB (lead company), Stone and Webster, Rolls-Royce (England) and AEA Technology (England) to develop a new generation reactor design (SPALWR – Small Passive Advanced Light Water Reactor) based on proven technology, while making the plant small (300 MWe) and more passive in nature.

- Completed a preliminary NSSS design (300 MWe) using innovative design concepts.
- Proposed the resultant design, SIR (Safe Integral Reactor) to the U.S. DOE for design and licensing (to U.S. NRC requirements).
- Arranged sponsors in the United Kingdom to fund the first plant build.

Combustion Engineering, Inc. **1980-1989**
<u>Regional Project Manager:</u> West, Northwest, and South Central Areas of the U.S.

Managed all activities associated with the startup, initial commercial operation, and subsequent services work and refueling operations for CE's most advanced nuclear power plants.

Arkansas Nuclear One-Unit 2, 900 MWe
Waterford-3 (Louisiana), 1100 MWe
San Onofre Nuclear Generating Station-Units 2,3 (California), 2200 MWe total
Palo Verde Units 1,2,3 (Arizona), 3600 MWe total

Managed all services work for non-CE plants in the same region. Principal customers included Pacific Gas and Electric, Portland General Electric, and South Texas Project.

- Led a major effort to improve reliability of the ANO-2 digital reactor monitoring and protective system – the first of its kind in the world. Result: ANO-2 now produces 10% more power with 10% less fuel than ANO-1, an identically sized plant designed by a competitor.
- Directed delivery of first high-density core; and directed major modifications during the first refueling outage.
- Staff included four site managers, and four project managers.
- These plants became the technology leaders for CE and ABB. Subsequent nuclear reactors sold to the Republic of Korea utilized this proven technology.
- Business volume: $65M/year

Combustion Engineering, Inc. **1980-1991**
<u>Regional Project Manager: Europe</u> (simultaneous with above)

Managed the fuel storage portion of Project CLAB, the world's first AFR (away-from-reactor) interim spent fuel storage facility. Contracted a subvendor, OMV of Ornskoldsvik, Sweden, for fabrication, and directed two site managers to manage field installation activities. Marketed reactor services to European reactors in Stockholm, Sweden, and Zurich, Switzerland; and managed resultant contracts for first-time applications of new technology. This effort introduced new worldwide services businesses for CE and ABB valued at over $1.5B.

- Completed the Project CLAB design on schedule, without any changes, and licensed the under the requirements of SKI (Swedish Nuclear Inspectorate). This was the first seismically qualified system in Sweden; and required analyses of many plant-specific accident scenarios. The facility went into operation in 1985 and has operated flawlessly. Equipment deliveries continued until 1990 without a single deviation.

- Developed advanced reactor steam generator remote inspection and repair equipment used under intense radioactive environment, under contract to the Swedish State Power Board. Support staff: 50 engineers from CE and SSPB.
- Developed the only incore monitoring system to be used at a non-CE plant, Ringhals-2, based on the technology developed at ANO-2. In combination with a changeout in steam generators, this permitted a 7-9% increase in power, which more than paid for the plant modifications.
- Provided specialized electrical cabling for accident monitoring under all potential accident conditions; and the first set of steam generator nozzle dams: Beznau, Switzerland.

Combustion Engineering, Inc. **1977-1979**
<u>**Assistant Project Manager**</u>

Responsible for assuring completion of all engineering development and fabrication of reactor components for two nuclear reactors, Arkansas Nuclear One – Unit 2 and Waterford-3, and obtaining U.S. NRC licenses.

- All components were completed and delivered on schedule, both plants.
- Licensing with NRC was completed on schedule, both plants.
- Directed a major testing and design change evolution that required modification of the nuclear fuel and reactor vessel internals (in-situ). These changes also required NRC approval.

Combustion Engineering, Inc. **1976-1977**
<u>**Assistant Project Manager**</u>

Directed all activities involving proposal development, sales, design, analysis and manufacturing specifications for a new CE product: high density spent fuel storage racks.

- Completed first high density spent fuel storage rack design in the U.S. This became a very important product for CE, as it made up for waning nuclear reactor sales; and a necessary product for utilities whose nuclear units would have been required to shut down due to lack of storage space resulting from President Carter's non-proliferation policies, i.e., halt to fuel reprocessing in the U.S.
- Directed state-of-the-art analysis techniques used to address high seismic applications using non-linear dynamic analyses. This subsequently became the industry standard.
- Average revenues: $15M/year

Combustion Engineering, Inc. (some overlap with above) **1975-1977**
<u>**Project Engineer**</u>

Coordinated all engineering activities within the Reactor Design, Mechanical Design, and Physics engineering departments.

- Facilitated completion of the Maine Yankee, Calvert Cliffs, Millstone 2, and St. Lucie 1 reactor designs.
- Personally completed the advanced fuel design for ANO-2 (see above).
- Facilitated the analyses and fabrication of all fuel reloads.
- Participated in establishing the next generation reactor design (System 80).

Combustion Engineering, Inc. **1970-1975**
Supervisor, Nuclear Fuel Design

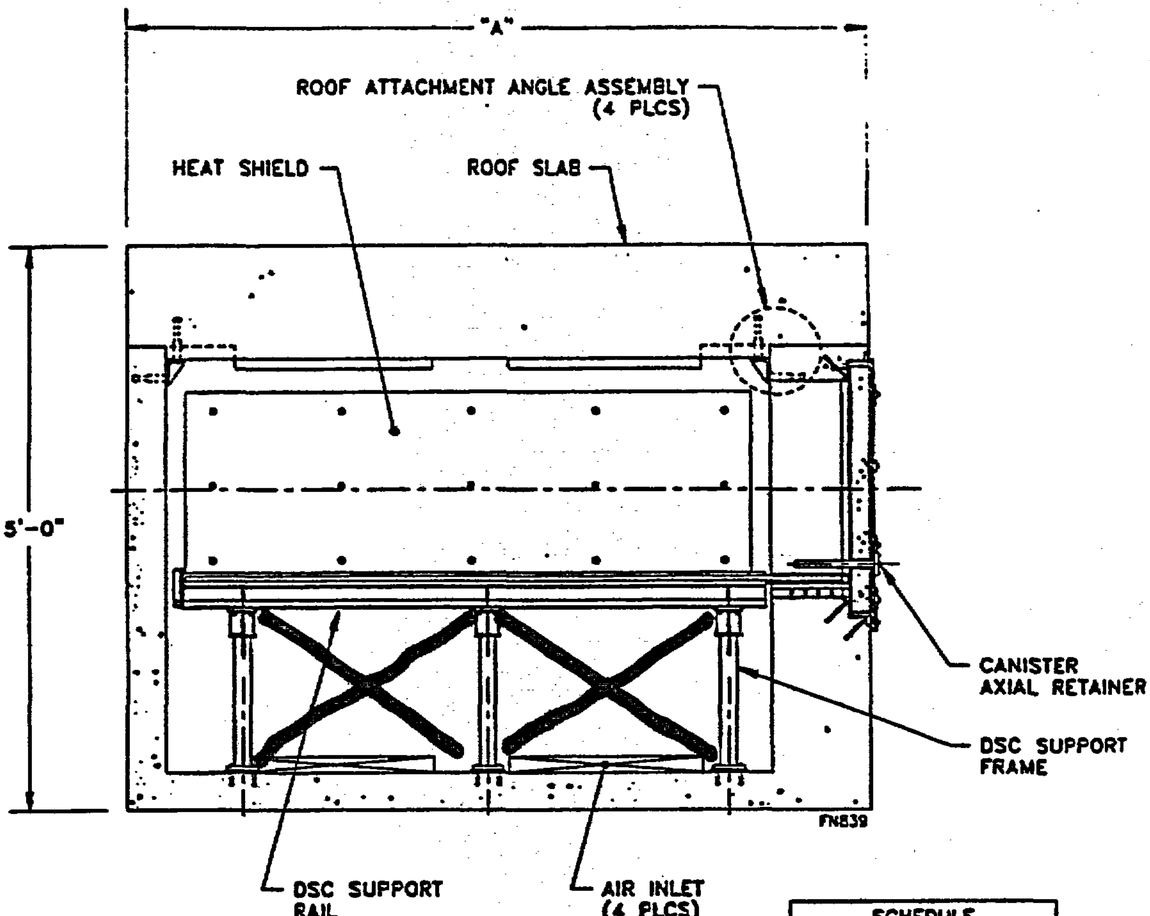
Lead responsibility for designing nuclear fuel and control rods for commercial reactors in the U.S.; for specifying all manufacturing requirements and assuring that they are met; for performing in-situ examinations of spent fuel; and for associated R&D programs.

- Inspected first CE irradiated fuel assembly (Palisades)
- Discovered irradiation induced growth of Zircaloy – the main structural component of CE's fuel assembly design. This required redesign of fuel and reactor vessel internals structures.
- Identified potential for poison rod growth (later confirmed at several reactors), requiring modifications of the rod designs.
- Led two campaigns to repair damaged fuel assemblies at nuclear reactors (Ft. Calhoun and Maine Yankee)
- Identified and corrected design flaws with the control rod drive mechanism attachment to the control rods which otherwise would have prevented full scram in reactor.
- Product value: $30M/year to CE; about ten times this amount to the utility which owned the contained uranium fuel.

APPENDIX B

MARKED PAGES FROM NUHOMS FSAR

(Total of six pages including this title page)



SCHEDULE	
FUEL TYPE	DIMENSION "A" MODULE LENGTH
PWR	19'-0"
BWR	19'-10"

Figure 4.2-6
<u>Prefabricated NUHOMS® Module Longitudinal Section</u>

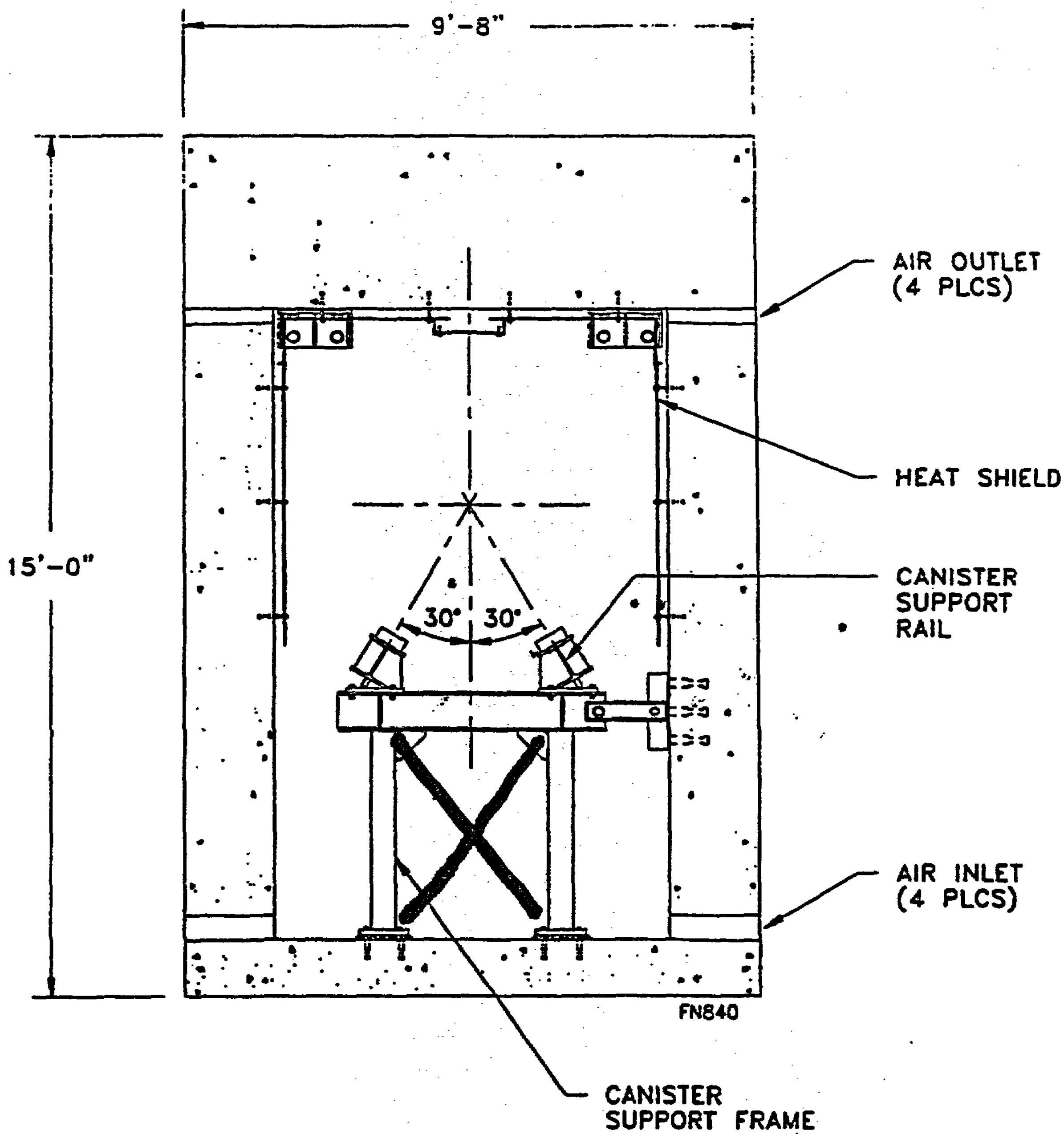


Figure 4.2-7
Prefabricated NUHOMS® Module Cross-Section

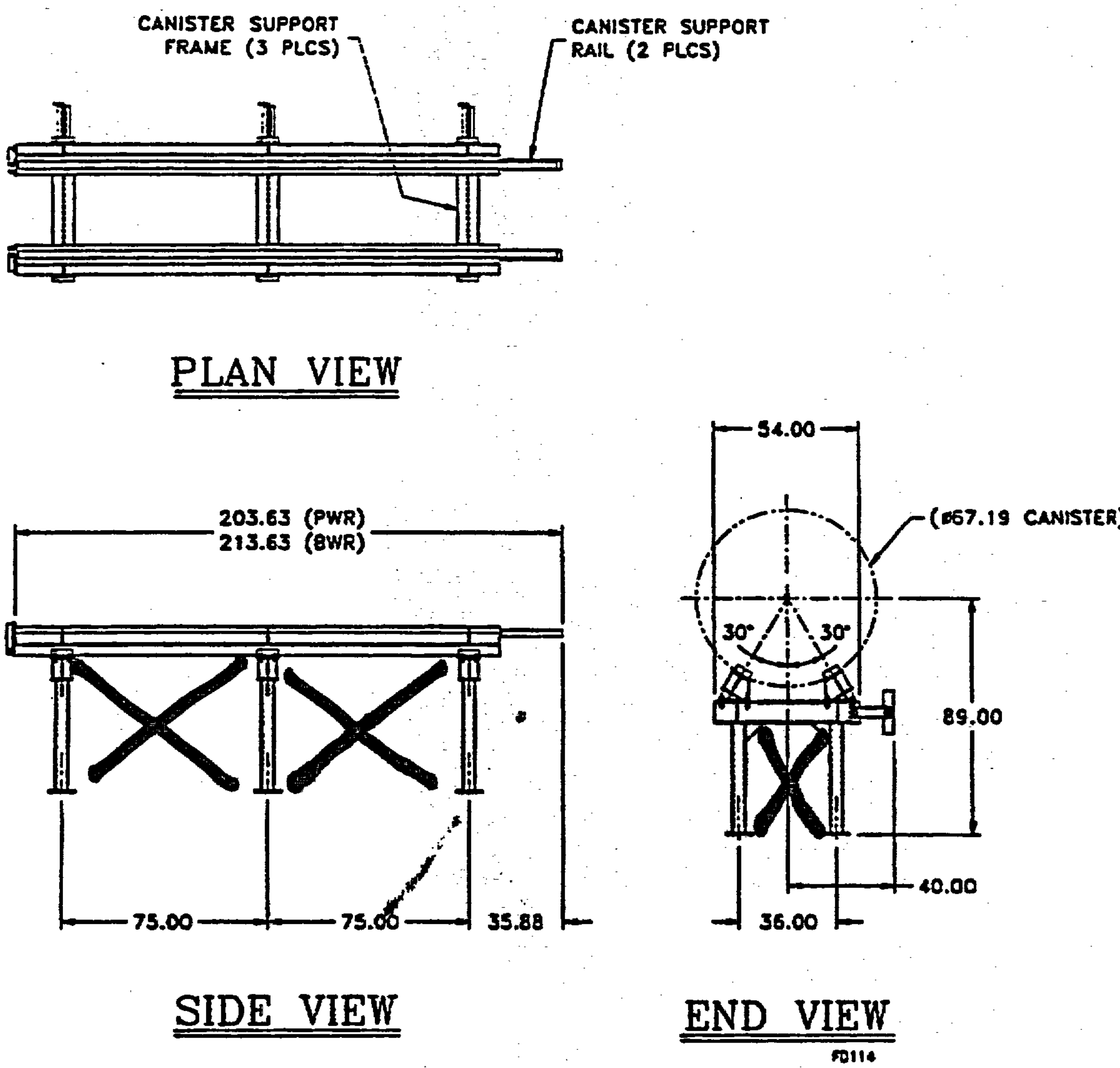
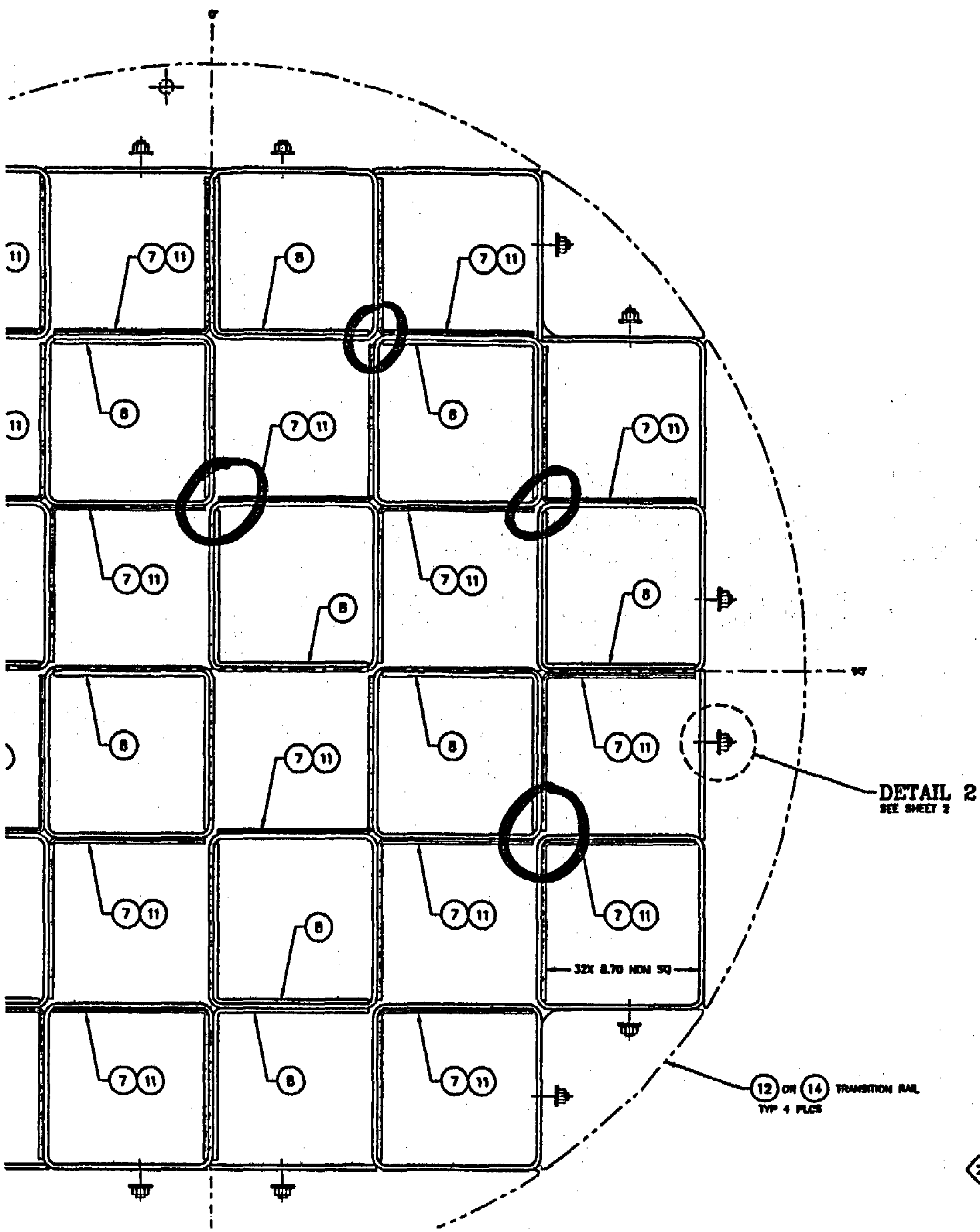


Figure 4.2-8
DSC Support Structure

EV	DESCRIPTION:	DRAWN:	ORIGINATED:	VERIFIED:	APPROV

CENSING:



NOTES:

1. (PROPRIETARY).

2. ◇ QUANTITY, SIZE, MATERIAL, AND LOCATION OF ITEMS 4, 5, 6, AND 9 AS REQUIRED. ALTERNATE DETAILS OR METHODS OF ATTACHMENT ARE ACCEPTABLE WITH TN WEST APPROVAL.

3. (PROPRIETARY).

4. (PROPRIETARY).

5. ◇ TUBES MAY BE EXTRUDED OR FORMED. ONE OR TWO LONGITUDINAL FULL PENETRATION WELD SEAMS ARE ALLOWED IN THE FLAT SIDES OF EACH FORMED TUBE. 100% VISUAL INSPECTION OF BOTH SIDES OF WELD SHALL BE PERFORMED. SEAMLESS OR WELDED AND DRAWN SLEEVES ARE ALSO ACCEPTABLE.

6. (PROPRIETARY).

7. (PROPRIETARY).

8. (PROPRIETARY).

9. ◇ CONNECTION IS ONLY TO FACILITATE FABRICATION/ASSEMBLY. ALTERNATE DETAILS OR METHODS OF ATTACHMENT ARE ACCEPTABLE WITH TN WEST APPROVAL. RAIL TO BE SLOTTED TO PERMIT UNRESTRAINED THERMAL EXPANSION.

10. (PROPRIETARY).

11. WELD SYMBOLS ARE PER ANSI/AWS 2.4-88. WELD SIZES ARE MINIMAL. ALTERNATE WELDS OF EQUIVALENT STRENGTH MAY BE USED WITH TN WEST APPROVAL.

12. CUTOUTS AT BOTTOM OF EACH FUEL COMPARTMENT CELL SHALL BE MADE TO FACILITATE DRAINING.

13. BASKET ASSEMBLY TO HAVE SUFFICIENT CLEARANCE WHEN INSERTED INTO CANISTER TO ALLOW FOR THERMAL EXPANSION.

14. (PROPRIETARY).

15. (PROPRIETARY).

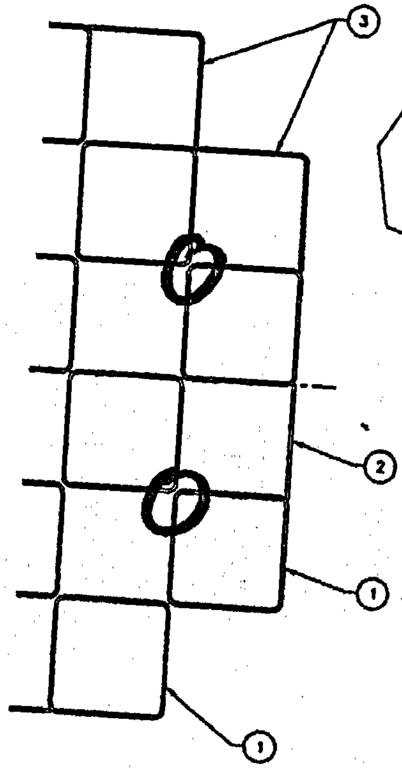
16. ◇ HOLES IN ITEMS 7, 8, AND 11 TO BE SLOTTED TO ALLOW THERMAL EXPANSION.

	15	4	NUH-32PT-1006	R90 ALUMINUM TRANSITION RAIL	--	A
	14	4	NUH-32PT-1006	R45 ALUMINUM TRANSITION RAIL	--	A
	13	4	NUH-32PT-1005	R90 SST TRANSITION RAIL	--	A
	12	4	NUH-32PT-1005	R45 SST TRANSITION RAIL	--	A
	11	20		ALUMINUM SHEET, .05 THK	TYPE 1100	B
	10	128		SHEET	ASTM A240 TYPE 304	A
◇2	9	A/R		SCREW, FLAT COUNTERSUNK HEAD CAP	SST	C
	8	12		ALUMINUM SHEET, .125 THK	TYPE 1100	B
	7	20		NEUTRON ABSORBING SHEET, .075 THK		B

EV	DESCRIPTION:	DRAWN:	ORIGINATED:	VERIFIED:	APPROVE

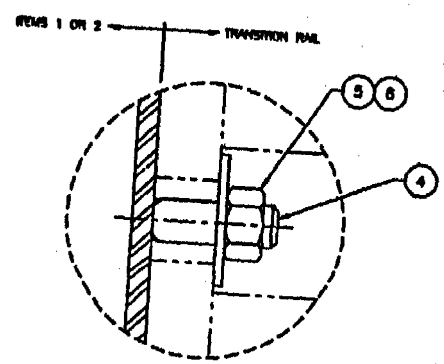
CENSING:

ETAILS



T MIDLENGTH
ONLY

PROPRIETARY



DETAIL 2